Radioactive waste management in Estonia: Milestone 1 Ready to make a knowledgeable commitment to a nuclear power programme

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Theoretical Background

Estonia is managing a Soviet nuclear waste heritage and radioactive waste from radioactive practices, mostly from medical sector and industry. Adapted legal and administrative frameworks are in place. Necessary financial and human resources are allocated.

Estonia now considers launching a new build (SMR) nuclear power programme. The report identifies additional responsibilities for radioactive waste that will come with a nuclear power programme and the options for safely and securely dealing with radioactive waste.

The objective of the present report is to contribute enabling the government and parliament of Estonia to fulfil IAEA milestone 1 and make a knowledgeable commitment to a new nuclear power programme, as recommended by IAEA [31].

By the end of the decade, we can expect the availability of SMRs on the market, as their designs continue to advance and evolve, although at present, there are no SMRs in operation and their designs remain proprietary. Light water SMR waste characteristics are comparable and close to existing large scale LWR waste.

SMR waste streams and the management and disposal of SMR nuclear waste are assessed based on available studies and the extensive international industrial experience with large scale LWRs.

Methodology

The present analysis follows the International Atomic Energy Agency (IAEA) guidance publication NG-G-3.1 (Rev.1) “Milestones in the Development of a National Infrastructure for Nuclear Power, chapter 3.17.1 “Ready to make a knowledgeable commitment to a nuclear power programme”.

The following tasks will be analysed or performed:

1) Existing capabilities, regulatory framework and experience with radioactive waste handling, storage, transport, and disposal in Estonia.

   The first task compiles Estonian publicly available information, evaluates and uses information collected during interviews in Estonia and at the occasion of expert presentations at various nuclear energy events in Estonia in 2022 and 2023. A paragraph is dedicated to the new regulatory framework if Estonia decides to start using nuclear energy

2) Additional volume of LLW and ILW, and the variety of isotopes expected from nuclear power facilities for the following reference reactors to the extent possible according to the data from publicly available sources.

   Task 2 is based on research of existing publicly available guidance (e.g., IAEA) and information (ANDRA, Orano, WNA) for the waste categories, the study of advanced assessments of SMR waste streams and the compilation of HLW isotopic data from international sources.

3) Technological options and research on the ultimate disposal of spent fuel and HLW from reprocessing.

   Task 3 presents the different available nuclear fuel cycle strategies, technological options and main advantages and disadvantages based on the author’s research, know-how and experience. An additional paragraph deals with plutonium and reprocessed uranium management opportunities and inconvenients.

4) Options for financing spent fuel and HLW management and disposal of spent fuel and financial aspects relating to the safety of radioactive waste management.

   Task 4 summarises the way of financing spent fuel and HLW management step by step based on IAEA’s recommendations. Based on the author’s research and know-how an additional sub-paragraph investigates the impact on the cost of a final disposal infrastructure when implementing an ‘open cycle’ versus a ‘closed cycle’ strategy.

5) The human resource and other infrastructure development needs associated with radioactive waste management for a nuclear power programme and the possibilities for their development in cooperation with other countries or international organizations.

   Task 5 assesses the potential competences needed and staffing aspects in the regulatory authority and waste agency and provides suggestions for cooperation with other

countries or international organizations.

6) Provide recommendations for the development of the national radioactive waste management policy.

   Task 6 presents a line-up for the national policy, i.e., the principles and goals for radioactive waste management (proposing three waste management possibilities) and describes the technical strategy, or strategies, for the management of their radioactive waste, i.e., the approaches for the implementation of the policy.

7) Key take-aways

   The Executive Summary paragraph includes a summary of backend HLW waste streams for all available strategic LWR nuclear fuel cycle options. At the end the author's recommendation for Estonia is provided.

Abbreviations and definitions

Small Modular Reactor (SMR)

Both the design and the construction of SMR can be “modular”. Modularity in design refers to multi-modules SMRs: one plant is made of several units. Units can be successively added to the plant at the initial construction time or later. Modularity in construction refers to the shipment of in-factory manufactured modules. This reduces on-site activities, thus the construction time and risks.

Light Water Reactors (LWR)

Over 80 % of the current installed commercial nuclear reactors are LWR. In LWR water is used as both coolant and neutron moderator. The neutron spectrum is thermal. LWR are divided between Pressurized Water Reactor (PWR) and Boiling Water Reactor (BWR)

Pressurized Water Reactor (PWR)

Among LWR, PWRs have pressurized water as coolant and moderator. Water flows through the primary circuit at very high pressure – usually 155 bars – into steam generators where a secondary circuit is connected. Water from the secondary circuit becomes steam when heated in the steam generator and drives the turbine. Examples include – but are not limited to – EPR, AP1000, VVER and Hualong reactors.

Boiling Water Reactor (BWR)

Among LWR, BWRs have a primary circuit directly connected with the turbine. Water – usually at 70 bars – is heated in the nuclear core into steam which drives the turbine. BWR-6, ABB-3, ABWR are examples of BWR designs.

A.L.A.R.A. – Estonian waste management agency

AMR – Advanced modular reactor

ANDRA – French national nuclear waste agency

BWR – Boiling Water Reactor

Cigéo – French Industrial Centre for Geological Disposal

COVRA – The Netherland's WMO

CSD-V – Universal vitrified standard canisters (for fission products)

CSD-C – Universal compacted standard canisters (for metallic waste)

DBD – Deep borehole disposal

D&D – Decommissioning and dismantling

DGR – Deep geological repository

DPC – Dual purpose cask

DSRS – Disused sealed radioactive sources

DU – Depleted uranium

EC – European Commission

EIA – Environmental Impact Assessment

ERDO – European Repository Development Organisation

EU – European Union

EUP – Enriched uranium product

Euratom – European Atomic Energy Community

ERU – Enriched reprocessed uranium

FBR – Fast-breeder reactors

FOAK – First of a Kind

FTE – Full time employee

HTG – High temperature gas cooled

HLW – High level waste

HM – Heavy metal

IAEA – International Atomic Energy Agency

IEA – International Energy Agency

IFNEC – International Framework for Nuclear Energy Cooperation

ILW – Intermediate level waste

KBS-3 – Swedish and Finish disposal concept

LL- Long lived

LLW – Low level waste

LILW – Low and intermediate level waste

LWR – Light water reactor

MOX – Mixed oxide

MSR – Molten Salt Reactor

NEWG – Nuclear Energy Working Group

NRC – US Nuclear Regulatory Commission

NPP – Nuclear power plant

NPT– Non-Proliferation Treaty

NRC – US Nuclear Regulatory Commission

OECD – Organisation for Economic Co-operation and Development

Posiva Oy - Finish nuclear waste management company

PWR – Pressurised Water Reactor

PUREX – Plutonium uranium reduction extraction

RADON – LLW and ILW storage facility enterprises established the USSR

RCCA – Rod cluster control assembly

R&D – Research and development

RepU – Reprocessed Uranium

SKB – Swedish Nuclear Fuel and Waste Management Company

SL – Short lived

SMR – Small Modular Reactor

SNF – Spent nuclear fuel

STENFO – Swiss “Stilllegungs- und Entsorgungsfonds”

STUK – Finnish Radiation and Nuclear Safety Authority

TSO – Technical support organisations

VLLW – Very low-level waste

WIPP – Waste Isolation Pilot Plant

WMO – Waste management operator/organization

WNA – World Nuclear Association

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Executive Summary & Key-Take-Aways

Estonia established a National Nuclear Energy Working Group (WG) in April 2021 to analyse the feasibility of using SMR-s in Estonia beyond 2030 to achieve its 2050 climate goals and ensure energy security. Estonian Government's expects the WG to present its final comprehensive report on 19 issues according to the IAEA Milestones Approach by December 2023.

The EU's Radioactive Waste and Spent Fuel Management Directive 2011/70/Euratom requires that all EU countries have a national policy for spent fuel and radioactive waste management and that they draw up and implement national programmes for the management of these materials.

Prior to developing a waste management strategy, the legal framework under which the waste management will operate needs to be established, together with the regulatory framework defining how the waste management activities shall be regulated. The government is responsible for establishing this framework and needs to designate an independent regulatory body to enforce the waste management regulations.

It is important to develop the waste management policy and strategy in consultation with all stakeholders in waste management.

From a multinational perspective, when formulating a national policy and strategy for radioactive waste and spent fuel management it will be important to address export/import of radioactive waste, spent fuel management, radioactive waste management, and public information and participation.

There are a variety of fuel cycle and waste management options: open vs. closed cycle, own vs. third party recycling, wet vs. dry interim storage, mined repositories vs. deep boreholes, etc. It is important to plan for flexibility, not to rule out any path at the beginning, unless it is unavoidable for ethical reasons. (Political reasoning can and probably will change over the NPP lifetime and beyond.)

Choosing an open or closed cycle policy or 'partially closed cycle' is a national strategic task.

The future new nuclear law shall establish the independent regulatory body having the authority to determine all matters relating to the control and supervision of the nuclear sector in relation to safety, security, and safeguards.

The nuclear law will also establish the legal framework under which the waste management will operate and establish and define the responsibilities of a national waste agency.

A fundamental prerequisite for implementing a sustainable waste disposal programme is the provision of financial resources to cover the costs of the programme including those that will arise after the NPP operating lifetime.

A qualified and capable workforce will be necessary for staffing of the future regulatory body and waste agency.

The present report Key-Take-Aways are summarised below together with an overview flow-chart of all HLW backend waste stream, waste funding and waste management options.

Fuel cycle key take-aways

Estimation of HLW waste volumes

The lifetime quantity of SNF of power reactors depends on several factors (like fuel type, burnup, efficiency, availability). Today's operating experience shows that a 1.000MW LWR will generate around 25t of spent fuel per year.

A variety of SMR designs are currently proposed. No civil operating experience is available yet and waste streams can only roughly be estimated based on the spare information available.

BWRX-300 is a small modular boiling water reactor presented by GE-Hitachi. The following are very rough estimations to give an order of magnitude:

The reactor core contains 240 fuel assemblies in fuel cannels. The expected lifetime of the plant is 60 years. During operation one reactor will produce approximately 2.400 spent fuel units (240 + 36 per year, in 24-months cycles) according to GE Hitatchi – see also paragraph 3.4.2.

The stated total mass of one fuel unit is 324kg[1]. The BWRX-300 will generate around 12t of spent fuel per year. Three such BWRX-300 reactors with 900MW capacity will produce approx. 35t of spent fuel yearly.

Let us assume the heavy metal weight of one BWR 10x10 assembly at ~200kg[2]. If all spent fuel assemblies (of three BWRX-300) of one year of operation were reprocessed about 15 vitrified HLW packages[3] would be produced (CSD-V package volume: 180 litres, package mass: 489 kg) and about 18 LL-ILW compacted metal packages[4] (CSD-C package volume: 180 litres, package mass: 700 kg). In addition, because of the BWR technology and depending on if the reprocessor takes them, ~7.200 fuel channels will potentially have to be managed as ILW.

Such reprocessing would also result in about 200 kg of separated total plutonium yearly (~1%), enough for 8-10 fresh MOX fuel assemblies. (MOX fuel is made up of up to 12% total plutonium, of which 4-5% is fissile, and depleted uranium.)

In a thought experiment, if all spent fuel assemblies of three BWRX-300 SMRs generated during 60 years of operation would be reprocessed, i.e., 3 x 2.400 = 7.200 fuel assemblies one could expect, depending i.a., on burnup between 2.900-3.350 CSD-V and around 3600 CSD-C. The corresponding total plutonium would amount to around 15t.

Operational and decommissioning waste (see paragraph 3.4.4)

Different types of low and intermediate radioactive waste are generated during reactor operation (e.g., filters, used components and contaminated industrial waste) and will be reduced in volume, conditioned, packaged, and stored prior to its disposal. This waste is mainly VLLW or LLW and can be temporarily stored at the reactor site.

D&D waste includes decontamination solutions and materials, contaminated building materials, pieces of metal or wood, electric wires, etc.

[1] According to [9]

[2] See Appendix 3, KK Mühleberg representative BWR 10x10 UO_2 spent fuel characteristics.

[3] Average production rate of 0,7-0,8 packages per t_{HM}.

[4] Average production rate of 0,65 packages per t_{HM}

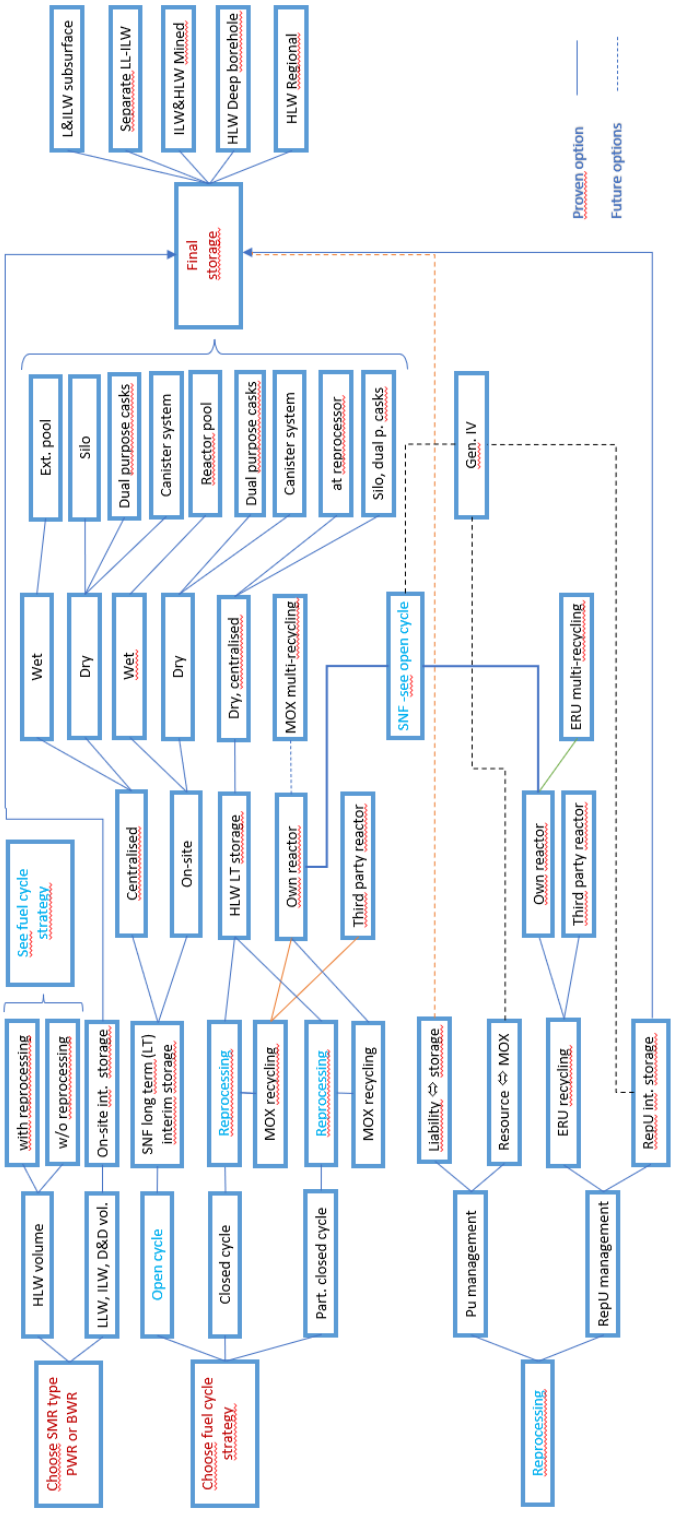
The waste volumes and associated management costs are difficult to estimate and depend on many boundary conditions. For example, allowing a release threshold may reduce volumes drastically. A typical 1000 MW(e) PWR or BWR produces between 5.000 and 10.000m^3 of decommissioning waste.

Looking at the Finish example[5], the costs of managing operational waste are expected to amount ~2% of the total spent fuel and radioactive waste management costs. The costs of decommissioning the plants are estimated at ~15%.

[5] A preliminary study of the National Institute of Chemical Physics and Biophysics is analysing the potential spent nuclear fuel disposal options for Estonia and associated cost forecast – see paragraph 4.2.2.

HLW waste streams

The flow-chart summarises the backend HLW waste streams for all available LWR nuclear fuel cycle options. By volume, HLW forms less than 1% of the global volume of radioactive waste, but it consists of about 95% of the total activity of the radioactive waste.

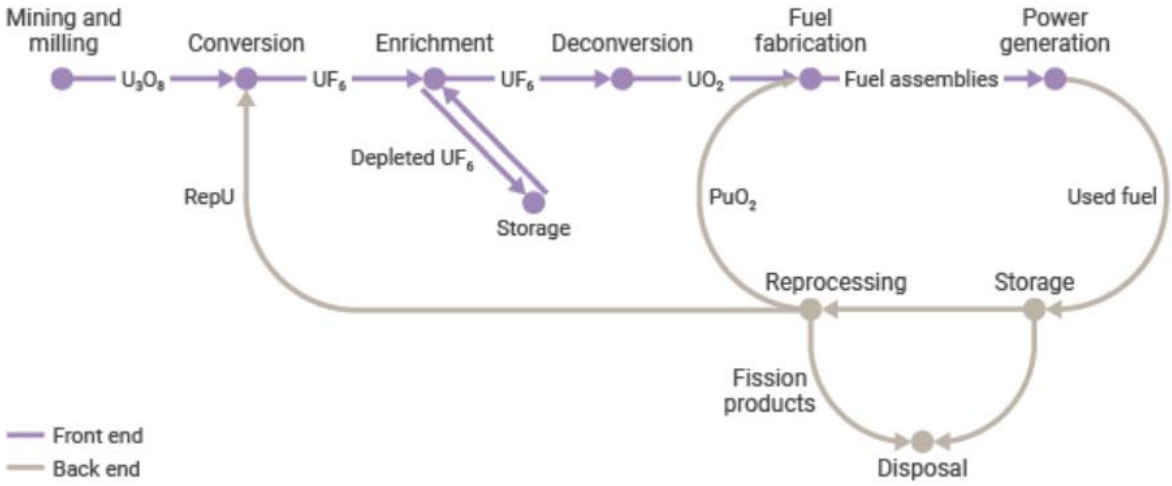


Fuel cycle strategy options (see paragraph 4.1)

The available strategy options ensure safe and cost-effective overall management of spent fuel and can be described as follows:

1) The 'open cycle', 'once through' or 'direct disposal' strategy, in which spent fuel is considered as waste,

2) The 'closed cycle' (including the 'partially closed cycle') strategy, in which spent fuel is considered to provide a potential future energy resource.



1) Open cycle

The 'open cycle' waste management route englobes many positive aspects. Among other early, short-term expenses are favourable and the option permits switching strategies also at a later moment in time. Several associated choices will have impact on later waste management options and costs.

First, the spent fuel must cool down for a few years in the reactor pool. From there it will go into long term interim storage, i.e., for several decades. Long term interim storage can be provided under wet (under water, in a pool) or dry (usually in casks) conditions.

a. Wet storage presents the advantage that the spent fuel is always accessible for inspection and repair. If the reactor pool's capacity is designed to accept all spent fuel assemblies accrued over the reactor lifetime, then the spent fuel assemblies can easily be stored there until after reactor shutdown and must be moved away only at NPP decommissioning.

   If the reactor pool's capacity is not designed for lifetime storage the spent fuel assemblies shall be evacuated in due time to avoid pool saturation[6]. The operator may stick to the wet storage option and place the spent fuel assemblies in an external pool. On-site transfer to such external pool can be provided with a shuttle or, if adjacent via under-water transfer. Off-site storage is usually centralised, i.e., will accept spent fuel from more reactors and implies available spent fuel transport solutions.

[6] BWRX-300 spent fuel pool provides a capacity of 8 years of full-power operation.

b. Alternatively, after initial cool-down in the reactor pool the operator may decide for continued dry storage of the spent fuel in storage and transport casks (i.e., dual purpose) or canister systems. Dry storage may be provided on-site or off-site. Any dry storage solution includes available means of transportation.

   Dry storage is the preferred solution for avoiding reactor pool saturation and easing its management. It requires relatively reduced capital and operational expenditures.

   The advantages and inconveniences are described in paragraph 4.1.1.1 “Open cycle: facts, practices and options”.

As a bottom line it can be said that dry and wet interim storage are technically sound, mature, and safe waste management solutions and represent immediate cost-efficient options that delay most of the waste management expenses until final storage.

In all above-described cases (wet or dry) after a period of interim storage, the spent fuel will be repacked, encapsulated in a robust, corrosion resistant container to meet disposal acceptance criteria and will be disposed of in a deep geological repository.

Spent fuel assemblies contain a few kilograms of fissile uranium and plutonium. Anti-proliferation safeguards, surveillance and special protection measures are mandatory at all times.

Today most countries and operators using civil nuclear power practise an open cycle strategy.

2) Closed cycle

In reprocessing spent fuel is separated into several main components: uranium, plutonium and HLW (containing minor actinides, fission, and activation products) and LL-ILW (containing activated metals).

Closed cycle implies reprocessing of spent fuel assemblies which then facilitates intermediate storage of HLW and its final disposal by reducing the waste volume by a factor of 5 and its toxicity by a factor of 10 (as compared to the open cycle route). Further it allows fuel recycling of separated uranium and plutonium, saving natural uranium resources. The advantages and inconveniences are described in paragraph 4.1.1.2 “Closed cycle: facts, practices and options”.

a. Reprocessing: after initial cool-down in the reactor pool (which may be shorter as in the open cycle) the spent fuel is transported (abroad) and reprocessed to recover valuable fissile materials (uranium and plutonium). Long-term storage of fission products requires the stabilization of the waste into a form that will neither react nor degrade for extended periods. This is obtained through vitrification and conditioning in stainless steel containers, placed first in intermediate storage (dry, shaft-type or in dual purpose casks) to allow the decay heat to be reduced before final disposal. Conditioned HLW packages potentially already meet final disposal acceptance criteria.

   Deep geological repositories for HLW from reprocessing will have a smaller footprint. Final disposal is optimized with standardized types of waste and will probably be straight forward, i.e., not necessitate repacking and conditioning and potentially be less expensive.

b. Recycling: uranium and plutonium can be recycled as “fresh” nuclear fuel for reactors increasing energy security by potentially saving up to 30% natural uranium.

   A “moxable” reactor, own or third party, must be provided for MOX recycling. Light reactor adaptations will be necessary: reinforcement of control rods pattern, increase of boron

concentration in boron tanks and operational adaptations needed for the receipt of fresh fuel. More details are given in paragraph 4.1.1.3 “Plutonium recycling”[7].

MOX fuel fabrication and logistics are more complex and more expensive than for UO_2 fuel.

It shall be noted that time span of about 15+ years will pass between SNF reactor unloading and fresh MOX fuel reinsertion/reactor use[8]. Eventually when the reactor is shut down the remainder of spent fuel may be reprocessed (and benefit of HLW conditioning) but not recycled unless another fleet reactor or third-party reactor takes over the recycling task.

Special attention is required for plutonium management.

Reprocessing and recycling are industrially mature and sustainable options. Opting for a closed cycle strategy will require earlier cost expenditures for reprocessing. On the long term it will substantially reduce financial risks associated with HLW management.

Politically however, “first timers[9]” shall consider when engaging over the lifetime of their nuclear power program with non-national vendors. Historically friendly, allied nations shall be preferred.

When choosing a proven fuel cycle strategy (i.e., of current industrial practice) three high-level objectives should be considered:

- Reduction of the resource utilisation,
- Minimisation of the produced nuclear high-level waste (also beneficial to a repository program) and
- Economic competitiveness.

Both open and closed fuel cycle options and their variants (see “All options” flow chart) are practiced on large scale, technically proven and without major risks. The economic competitiveness depends mainly on two factors: the enriched uranium price and reprocessing/recycling costs.

[7] See details in paragraph 4.1.1.4 for “Uranium recycling”.
[8] Spent fuel into the reactor pool, initial cool-down of 3-5 years, transport to the reprocessing site, spent fuel reprocessing and Pu-separation, Pu-transport to the MOX fuel factory, fresh MOX fuel manufacturing, its delivery at the SMR site.
[9] Little nuclear experience, limited / no national regulatory system.

Final disposal options (see paragraph 4.2)

A deep geological repository is the most complex waste management facility to implement.

Disposal of spent fuel or HLW from reprocessing is provided in deep underground repositories in stable geological formations. Isolation is provided by a combination of engineered and natural barriers (rock, salt, clay) to prevent the radionuclides from reaching humans and the environment.

Several spent fuel or HLW disposal options are available or envisaged:

1) Minded repositories
2) Deep borehole disposal (concept)
3) Regional or multi-national repositories (concept)

1) The most widely proposed deep geological disposal concept is for a mined repository comprising tunnels or caverns into which packaged waste will be placed – see paragraph 4.2.1. "Mined repositories". Mined repositories are currently the only available, developed, and licensed final disposal solution.

   Excavation of a deep underground repository using standard mining or civil engineering technology is limited to accessible locations (e.g., under land or nearshore), to rock units that are reasonably stable and without major groundwater flow, and to depths of between 250m and 1000m. The contents of the repository should be retrievable in the short term, and if desired, longer-term.

   Mined repositories for HLW from reprocessing will have a smaller footprint and will probably not require sophisticated repacking and additional conditioning as is necessary for spent fuel assemblies.

   One major challenge associated with mined repositories relates to the future transport of wastes between the interim storage sites and the repository site.

2) The deep borehole disposal concept consists of drilling a borehole into basement rock to a depth of up to about 5000 metres, emplacing waste canisters containing used nuclear fuel or vitrified radioactive waste from reprocessing in the lower 2000 metres of the borehole and sealing the upper 3000 metres of the borehole with materials such as bentonite, asphalt, or concrete. At such depths the host rock is assumed to provide greater isolation and containment than for shallower mined repositories.

   The DBD concept is adaptable for both spent fuel assemblies and HLW packages.

   Boreholes can be drilled offshore as well as onshore in both crystalline and sedimentary host rocks.

   DBD reduces or eliminates the transport issue through its potential for on-site, dispersed disposal. The footprint for a multi-borehole array is quite small.

   DBD may allow relatively early disposal of heat-generating wastes shortening interim storage times.

   A large total volume, large package sizes or the need for retrievability potentially render the waste unsuitable for deep borehole disposal. The main disadvantage of DBD is the near irretrievability of the wastes. On the other hand, regarding non-proliferation safeguards and the

security of fissile materials, DBD could be advantageous, as recovery of packages would not be possible.

It shall be noted that advances in deep drilling technology over the past 20–30 years have led to the reconsideration of deep borehole disposal. The concept is, however not yet demonstrated and not licensed.

Preliminary comparative cost estimation let expect that for one 300MW SMR borehole disposal is noticeably cheaper than the mined deep geological repository option (400M€ vs.700M€ overnight). Larger amounts of waste reduce cost differences between the solutions and then make the deep geological repository cheaper (for 3 or more such SMRs).

The available options are described in paragraph 4.2.2. “Deep borehole disposal”.

3) The 2011 EC Waste Directive allows that two or more member states can agree to use a final repository in one of them. The export of radioactive waste for disposal in countries outside the EU is also allowed but only under strict conditions and binding IAEA standards.

   By law, some states prohibit import from and/or disposal of waste in foreign countries. Interested countries should harmonise their policies regarding nuclear waste import and export.

   Thus, a willing host country could accept (e.g., in return of substantial financing) waste from other countries in a national repository operating within the host country, or a fully international facility owned by a private company operated by a consortium of nations or even an international organization.

   Such DGR partnering approach drives down risk and cost (of all mega projects) and is the route to increased certainty.

   ERDO is a multinational working group established to study the feasibility of one or more shared geological repositories in Europe. A shared regional repository located in one Baltic country could also be investigated. Owning or managing only ‘closed cycle’, i.e., vitrified HLW canisters from reprocessing may be a favourable plus point in such context.

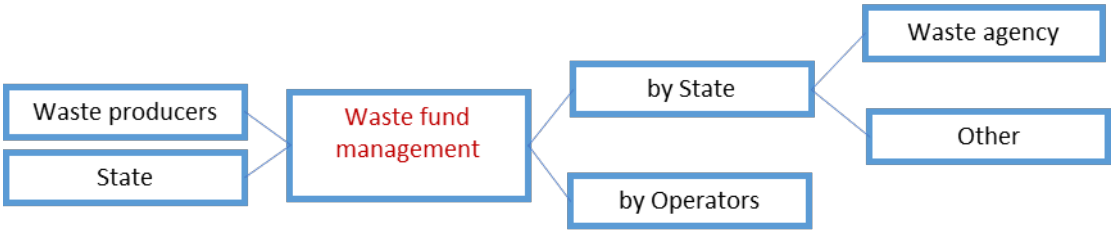
   However, despite intensive international efforts, no "voluntary" EU state has yet come forward.

   For more details see paragraph 4.2.3 “Multinational repositories”.

Final geological disposal is a distant obligation, especially in the case of nuclear “first timers”. Preparing for it on the national territory while in parallel searching for a regional solution (dual track approach) may be a positive and reassuring approach. Longer periods of safe HLW intermediate storage allow added decay heat reduction (an important DGR design factor) and accumulation of sufficient waste management funds.

Provision and management of financial resources (see paragraph 5)

A fundamental prerequisite for implementing a sustainable waste disposal programme is the provision of financial resources to cover the costs of the programme including those that will arise after the NPP operating lifetime.

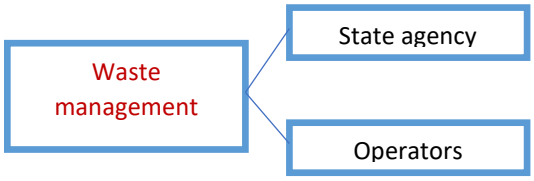


The most common funding sources are the waste producers and the State. Suitable funding and fund management mechanisms are presented in paragraph 5.1 “Financing spent fuel and HLW management”.

Implementation of an ‘open cycle’ or ‘closed cycle’ strategy will have an impact on the cost of a final disposal infrastructure. Under some circumstances it may also affect the interim storage necessities. Paragraph 5.2 compares the ‘open cycle’ and ‘closed cycle’ impact.

Modelling the two methods over long periods is unconclusive from a financial point of view. Overnight cost simulations hold the balance or incline to one option or the other depending on the respective study presumptions. Long-term investigation results are mainly driven by future risk assumptions.

The waste management task may be assigned to the operators or preferably a dedicated waste agency.



The main steps to be achieved in a first stage are: the nuclear energy policy, the funding development, the partner/vendor choice (considering geo-politics), the regulatory system, the planning to set process in place.

Suggestion for Estonia (see also attached flow-chart)

Estonia should keep all fuel cycle management options open, decide its strategy based on scientific knowledge when it comes to nuclear energy, separated spent fuel materials or HLW.

The nuclear policy should define spent fuel as a potential resource. If reuse of its materials is conceivable it should not be considered waste. After reactor use spent fuel should be stored in such way to allow transport to a reprocessing facility. Such intermediate storage may take place in the reactor pool if sufficient pool storage capacity is provided. Alternatively on-site dry intermediate storage of spent fuel may be provided in dual purpose (transport and storage) casks.

The SMR provider and the future operator should both be encouraged to foresee technically the use of MOX fuel from the very beginning. Only minor adaptations are required (see paragraph 4.1.1.3). Later adaptations are also feasible but associated with additional expenditures and licensing. Being capable to use MOX in a reactor doesn't imply that MOX must be used. Potential use of MOX is a valuable feature.

If Estonia chooses the 'closed cycle' path it should seek a partnership with a European reprocessing facility and also a MOX fuel provider. Costly and complex oversees transport of nuclear materials should be avoided. As of today, France is the only acceptable option. If the use of MOX fuel in Estonian SMRs is not desired a third party should be identified that is ready to burn Estonian plutonium. In such case the economic competitiveness of burning Estonian plutonium vs. burning market price EUP will be decisive. Usually, such arrangements go along with a financial subsidy closing the "MOX price gap".

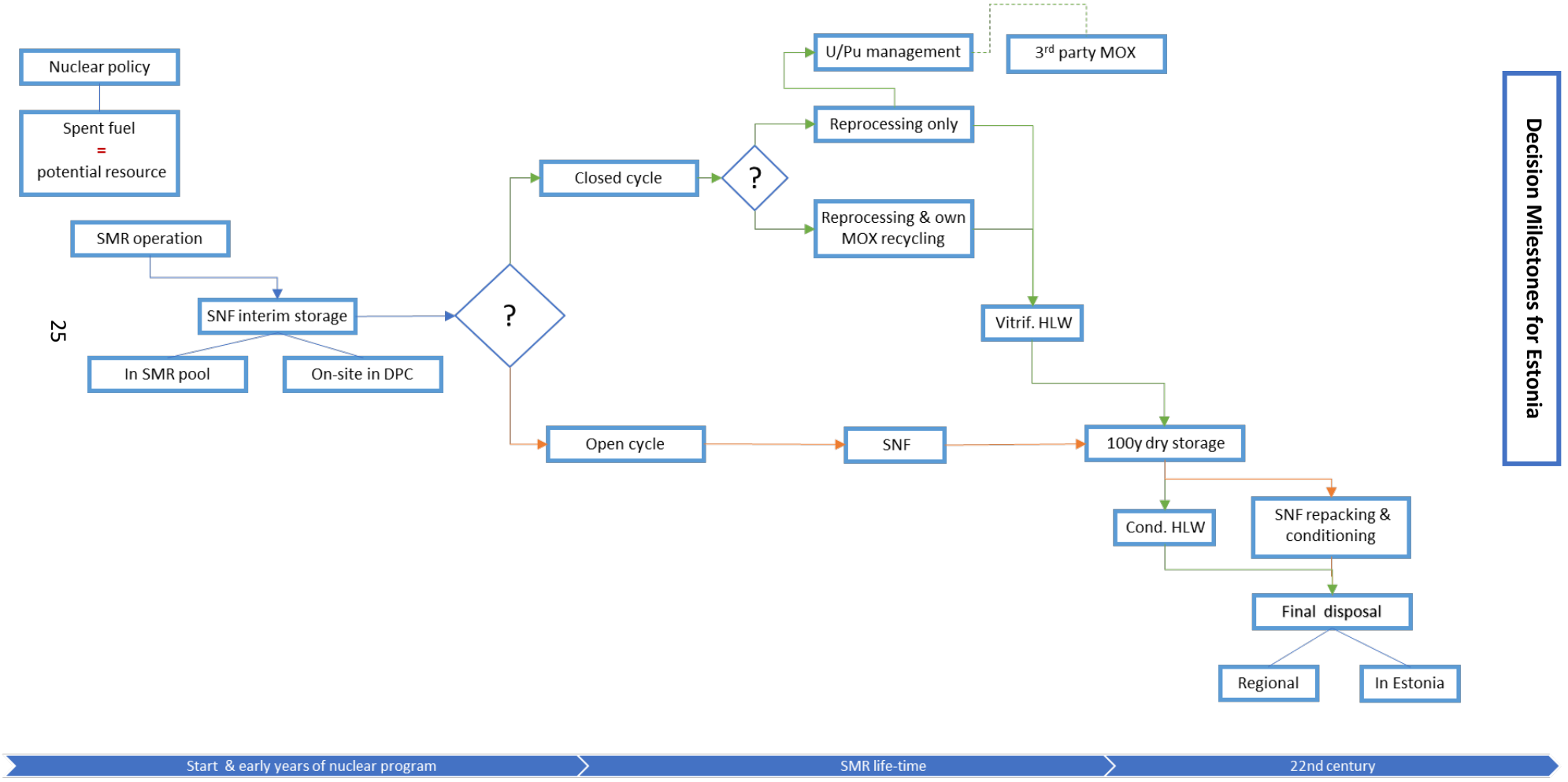
If the above recycling path is not practical the accumulated used fuel will be waste for direct disposal (open cycle)[10].

The state shall create the legal framework and oversee waste fund collection (polluter pays principle), its management and waste final disposal (via a dedicated waste agency). It will be challenging with a small SMR fleet to accumulate sufficient funding during the operational lifetime.

A dual track approach is advisable regardless of the fuel cycle management option.

The Dutch model could be an example. Covra, the Dutch waste agency, takes over and stores vitrified and conditioned HLW for 100 years before its final disposal, domestic or regional.

[10] On-site intermediate storage (dry or wet) of spent fuel as well as reprocessing and MOX recycling are operational expenditures and will be covered by the utility.


Nuclear policy
Spent fuel
=
potential resource
SMR operation
SNF interim storage
In SMR pool
On-site in DPC
?
Closed cycle
?
U/Pu management
3rd party MOX
Reprocessing only
Reprocessing & own MOX recycling
Vitrif. HLW
Open cycle
SNF
100y dry storage
Cond. HLW
SNF repacking & conditioning
Final disposal
Regional
In Estonia
Decision Milestones for Estonia
Start & early years of nuclear program
SMR life-time
22nd century

1 Introduction

In order to increase Estonia's energy security, sustainability and competitiveness and achieve its 2050 climate goals, "Estonia's 2030 National Energy and Climate Plan" and "Analysis of the opportunities to increase climate ambition in Estonia" have proposed the introduction of nuclear energy after 2030 as one of the possible solutions. This is one of the possible solutions for producing climate-neutral electricity in Estonia.

For Estonia, small modular reactors (SMRs) of Generation III+ or IV have been considered conditionally suitable. Small modular reactors are defined as nuclear reactors with a power output of up to 300 MW. They are smaller than conventional reactors and, theoretically, can be produced in a factory and assembled at a planned location. SMRs offer several advantages like advanced safety features, flexible power generation, smaller grid sizes compatibility, smaller footprints, flexibility in siting or better economic affordability[11]. However, in recent years some technical issues still attract considerable attention, one of it being the back-end solutions for the fuel cycle.

Waste disposal, especially of high-level waste (HLW), i.e., mainly spent fuel assemblies which contain 95% of the future radioactivity may seem a very distant task in the framework of a barely tackled new build programme. The lion's share of expenses for their management and disposal will occur in a distant future, however at a time when the nuclear power plants (NPPs) that generated them will probably be decommissioned, maybe even dismantled. For this reason, the necessary waste management funds as well as decommissioning and dismantling (D&D) funds must be generated during the NPP's operational lifetime and managed accordingly.

In contrary cases or when such decisions are delayed (as was the case with early nuclear power programmes) the burden falls to the state and taxpayer. Consequently, any new nuclear build programme today will require from the beginning to establish and secure waste management liabilities.

The present report will address the options of managing SMR waste and spent fuels following the International Atomic Energy Agency (IAEA) guidance publication NG-G-3.1 (Rev.1) "Milestones in the Development of a National Infrastructure for Nuclear Power, chapter 3.17.1". It identifies additional responsibilities for radioactive waste that will come with a nuclear power programme and the available options for safely and securely dealing with radioactive waste.

[11] Although SMRs are designed for lower upfront capital cost per unit, their economic competitiveness is still to be proven.

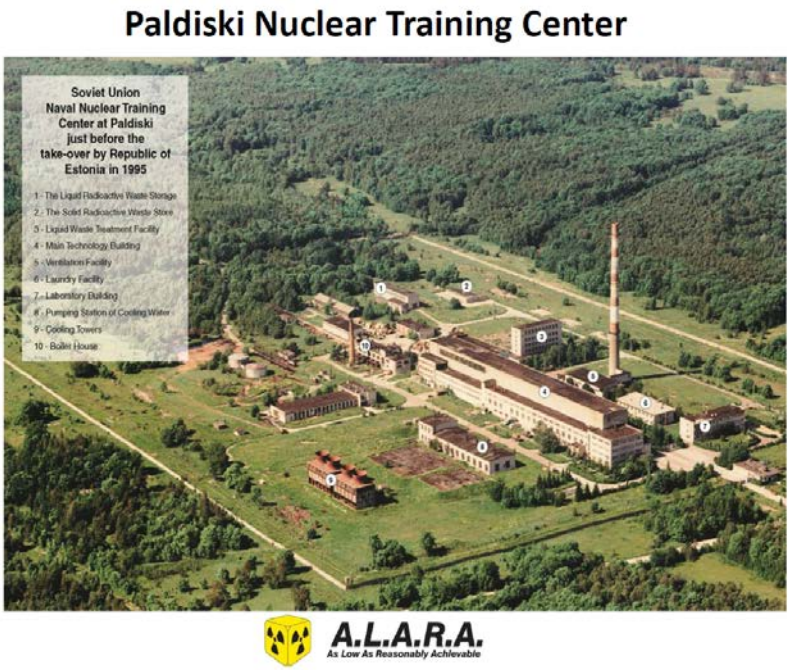
2 Existing capabilities, regulatory framework and experience with radioactive waste handling, storage, transport and disposal in Estonia

2.1 Existing Capabilities and experience with radioactive waste handling, storage, transport and disposal in Estonia

The sources of radioactive waste in Estonia are:

- Waste generated from first and second decommissioning stage of Paldiski site – see below,
- Waste arising from decommissioning of historical Tammiku disposal site (institutional waste) – 2008…2012, and
- Institutional waste (up to 5 m^3/a).

The Republic of Estonia has undertaken to decommission the training centre built on the northern tip of the Pakri Peninsula for the training of submarines of the former Soviet Union's Northern and Pacific fleets. Two nuclear reactors were in use in the training centre complex. The first nuclear reactor was built with a thermal capacity of 70 MW and the second reactor with 90 MW. Both reactors were shut down in 1989, and during the handover of the nuclear site, initial waste management work was carried out and nuclear fuel was removed from both reactors.



In 1994 the reactors were defueled, and the spent nuclear fuel was shipped to Russia. Nonradioactive components of the training stands were dismantled, hull sections housing reactor vessels with their primary circuits, auxiliary equipment and some additional wastes were partly grouted, seal-welded and enclosed into concrete sarcophagi.

Estonia agreed to take full custody for the site and future decommissioning of both reactor components and waste facilities together with the waste generated during the site operations. Ownership was transferred to Estonia in 1995. Since then, the work on monitoring, dismantling, decommissioning, decontamination and dismantling of the Paldiski facilities is in progress. Site is under administration of the Estonian Radioactive Waste Management Agency A.L.A.R.A. Ltd.

Today, an interim storage facility for radioactive waste has been built on the nuclear site, which stores the annual institutional radioactive waste (~5m^3/year) as well as the two reactor compartments around which sarcophagi were built.

In Tammiku the RADON-type storage facility for institutional waste with a capacity 200 m^3 was commissioned in 1963. By 1995, approx. 55% of the total volume (200 m^3) of Tammiku´s solid waste storage vault was filled with unpacked and unsorted waste. The site was finally closed 1995, and it was transferred to state company AS A.L.A.R.A. The storage vaults were covered with concrete slabs and a soil layer. In year 2006-2007 the environmental impact assessment was conducted to identify options to decommission the storage vault. The following option was chosen, retrieve the waste from the vault, decontaminate and dismantle the vault in order to release the facility site from regulatory control. The final end of decommissioning is green field which requires dismantling of the vault and removal of all building rubble. The site is under decommissioning since 2008:

- 2008-2011 – licence to practise radiation for the retrieval of radioactive waste from the Tammiku radioactive waste repository and transportation of radioactive waste to the Paldiski radioactive waste interim storage site for further management. By 2011, all radioactive waste had been removed, sorted, packaged and transported to the Paldiski site for further handling and storage.
- 2012-2017 – radiation practice license for radiological characterisation and decontamination of the facility.
- 2018-2023 – radiation practice license for final decontamination and demolishing of concrete structures of the facility and turning the site into green field. The concrete structures of the facility are demolished and transported to the Paldiski site. As part of the application process to request the release of the Tammiku site territory from regulatory control, the operator is currently preparing the radiation safety assessment report for the territory of the Tammiku site in order to assess its compliance with clearance criteria.

Paldiski interim storage facility was created in 1997.The Paldiski interim storage is in the Main Technological Building containing two sections for LILW capable housing 360 waste packages. The two sections host concrete and metallic containers with external dimensions 1,2x1,2x1,2 m, both with conditioned and unconditioned waste. Further, low active unconditioned waste (concrete rubble, contaminated metal) is packaged into ISO containers and 200 l drums. A radio-operated bridge crane is equipped with a specially designed gripper, for manoeuvring the waste packages.

2.2 Waste disposal – current repository concept

The disposal volume consists of the following waste types:

- ILW-SL (half-life up to 31 y) – 2100 m^3
- LILW-LL (half-life > 31 y) – 900 m^3

The reactor vessels and internals are classified as intermediate level waste based on Estonian legislation (heat generation less than 2 kW/m^3). As spent fuel was sent back to Russia there are no high-level waste present in Estonia.

Preferable option for decommissioning is full dismantling (without cutting of reactor vessel) with cutting into small pieces and fitting waste in standard concrete containers.

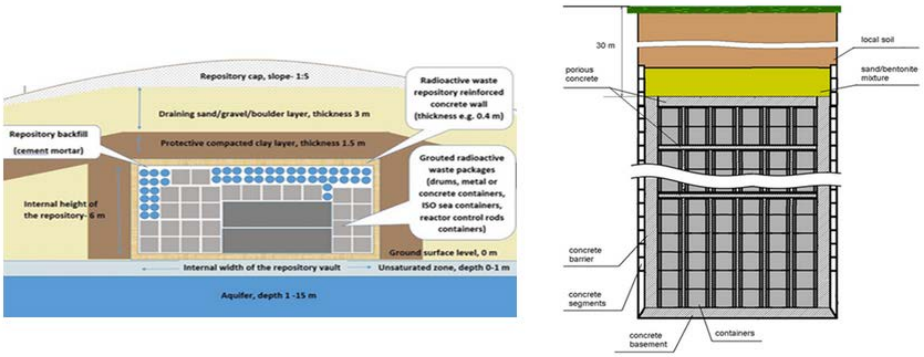
The designed life of the sarcophagi ends in 2045. Currently it is estimated that there is a need to establish a final disposal site for the existing waste.

Paldiski site today

The envisaged repositories concept is a near surface facility for the ILW-SL waste and shaft-type intermediate depth facility (60-80 m) in clay formation for the LILW-LL waste.

Repositories concept

- Near-surface facility;
- Shaft-type intermediate depth facility – clay formation, depth 60-80 m.



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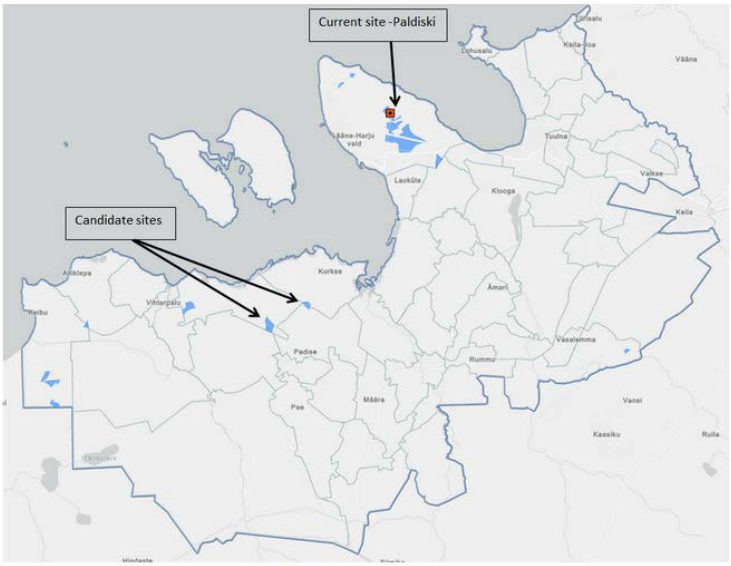
After negotiations the Lääne-Harju local county voluntarily agreed on 06.06.2019 to host the repository and be ready to start the local spatial planning process for such repository. This approach was accepted on Government Cabinet meeting on 04.07.2019.

It is worthwhile noting that locally worries are expressed that the Paldiski site may become a repository also for waste arising from a potential new nuclear programme.

The decision to build the disposal was made in 2016 without considering the nuclear power programme, which was not an issue at the time. The project of establishing disposal is ongoing and will continue despite decision will Estonia build nuclear power plant or not. It means that the planned disposal facility capacity is intended to accept only waste from current waste streams (Reactor compartment decommissioning, interim storage waste, and institutional radioactive waste) and not waste from nuclear power plants.

The currently envisaged site selection approach during 2021-2022 considers three possible locations in the Lääne-Harju County based on existing geological and socio-economical information. During this first stage 2 additional candidate sites to current site in Paldiski were identified. During public display 32 questions/recommendations from 5 institutions were received but no comments from private persons. A public hearing was held on 10.10.2022 with no resistance for none of the 3 selected sites.

Ranking results



A.L.A.R.A.
As Low As Reasonably Achievable

Studies in selected sites are started, drilling of wells is finished, and drilling core samples were taken.

Technical studies for the 3 selected sites will continue during 2022-2023 and be followed by detailed studies in 2024-2025 and targeting the repositories commissioning and operations by 2040.

For the construction of the final repository, a preliminary study was conducted between 2014 and 2015, during which a preliminary plan for the construction of a suitable final disposal site and the decommissioning of the reactor components was submitted to the Republic of Estonia.

During the preliminary study, the potential cost items of the submitted plan were also evaluated. (UAB EKSORTUS and SOSNY, 2015)

The preliminary studies are followed by the preparation for the decommissioning of the reactor components. The construction of the final disposal site needs to be completed by 2040 at the latest, in order to ensure sufficient time for the decommissioning of the reactor components. The planned operating period of the final disposal site is 2040-2060, and the closure will be carried out between 2060-2065.

Based on this schedule, a new study was conducted in 2022 by the Institute of Chemical and Biological Physics to update the costs assessed during the preliminary study, to supplement them with the costs of closure and post-closure activities, and to perform a risk analysis.

Based on the results of the analysis, several potential final disposal sites and decommissioning costs financing schemes are proposed to the Republic of Estonia, which would be compatible with the requirements specified in the "2011/70/EURATOM" directive (Council of the European Union, 2011), considering international practice, the specificity of Estonia's radioactive waste and its generation. This analysis serves as an input for updating the currently valid "National Action Plan for Radioactive Waste Management".

In summary the study found that with an 80% probability, the total cost of the basic scenario of the project as of today will be less than 105 million euros. Compared to 2015, project costs have increased by 25%. Due to the potential impact of inflation, the total cost of the project may increase up to 167 million euros in the future. The significant increase in the total cost due to inflation is largely more than ten years after the execution of the planned works. In order to ensure the coverage of necessary expenses (+ to avoid sanctions) and to reduce the potential impact of inflation, it is worth considering the establishment of a savings fund, the financing scheme of which will be analysed in the second stage of the study.

2.3 Legal and Regulatory Framework

2.3.1 Current situation

In Estonia, radiation safety activities are organised by the Ministry of the Environment within its area of competence through the Environmental Board by engaging other appropriate agencies and by taking inter alia into account field-specific operational experience, results of decision-making procedures, development of relevant technology and scientific research.

Radiation safety requirements are developed mainly in collaboration between the Ministry of the Environment through the Environmental Board, the Ministry of Social Affairs (Health Board, professional societies), the Ministry of Interior (Police and Border Guard Board, Rescue Board, Estonian Internal Security Service), the Ministry of Finance (Tax and Customs Board), the Ministry of Economic Affairs and Communications (radioactive waste management agency A.L.A.R.A. Ltd).

The current Estonian radioactive waste management policy is based on national legislative drafting and international principles. The policy and practice for radioactive waste management is to collect, characterize, manage, and store all Estonian radioactive waste under safe and secure conditions in dedicated facilities.

The Radiation Act provides measures to assess the national radiation protection system. The 10-years National Radiation Safety Development Plan provides objectives ensuring radiation protection and nuclear safety, radioactive waste management, responding to accidental and existing exposure situations, increasing radiation awareness and issues concerning natural and medical exposures.

The radioactive waste management agency's (A.L.A.R.A. Ltd) financing is assured via a yearly contract with the Ministry of Economic Affairs and Communications, based on an annual work plan (part of 4-years plan). In 2023 A.L.A.R.A. is employing around 10 people (maintenance, financial services, security guards, cleaning personnel etc.) with a budget of 653k€.

There are no nuclear power plants or facilities operating with nuclear fuel cycle in Estonia, neither any activities related to nuclear fuel cycle. Today Estonia's interest in nuclear safety is primarily related to the safety of nuclear installations in the neighbouring countries and to the implications that accidents at such installations, should they occur, may have on the health of the population and on the environment.

Legislation for a radiation protection framework was established in 1997, when the first Radiation Act entered into force. After joining the European Union, the process required preparation of several amendments to the Radiation Act, which were necessary to comply with the relevant legal framework of the European Atomic Energy Community (EURATOM).

In 2011 the requirements of the European Council Directive 2009/71/Euratom of 25 June 2009 establishing a community framework for the nuclear safety of nuclear installations were brought

into Estonian legislation by amending the Radiation Act. Based on discussion with the European Commission, the Paldiski site and a radioactive waste storage located on the same site do not directly fall within the scope of Directive 2009/71/Euratom. Therefore, Estonia must implement the requirements of this directive at a general level. The Radiation Act was amended with relevant definitions, requirements of passing a decision of principle by the Riigikogu (Parliament of Estonia) on establishment of a nuclear installation, obligations of the licence holder of nuclear installations and quality assurance requirements to ensure nuclear safety.

The Estonian Environmental Board, formed by merging the Environmental Board and the Environmental Inspectorate, has been the regulatory body since the 1st of January 2021. The new authority retains the same tasks and responsibilities as previous government authorities, such as authorization, review and assessment, inspection, and enforcement of radiation practices. The Radiation Act was amended to allow the consolidation of the two authorities into a single regulatory body.

The current regulatory body, the Estonian Environmental Board is employing 18 FTE out of which 13 are specialised in licensing, laboratory, and dosimetry aspects.

The Emergency Act was amended in July 2021 in order to improve the cooperation of the governmental authorities. The updated risk assessment of a nuclear or radiological emergency was adopted by the Environmental Board in January 2021, and the updated National Nuclear and Radiological Emergency Response Plan was adopted in February 2022.

Relevant nuclear legislation will be required, if Estonia decides to start using nuclear energy, and a new, independent regulatory authority shall be established to oversee all activities involving the use of nuclear energy or radioactive materials.

2.3.2 Nuclear legislation (required if Estonia decides to start using nuclear energy)

As mentioned previously, the "Estonian National Energy and Climate Plan 2030" and the "Analysis of Opportunities for Increasing Estonian Climate Ambition" have proposed nuclear energy after 2030 as one possible solution to increase Estonia's energy security and achieve climate goals. On November 5, 2020, Estonian Government adopted a decision to establish a Working Group of Nuclear Energy (NEWG) to define the nation’s positions towards the issue. NEWG was established by the Ministry of the Environment with the decree of the minister on the 20th of April 2021.

A comprehensive nuclear law shall be introduced that provides the legal framework for nuclear activities. It will regulate the peaceful use of nuclear energy aiming to protect humans and the environment.

The nuclear law’s scope will cover:

- Principles of nuclear safety (protection against ionizing radiation dangers, physical protection, non-proliferation, alignment with IAEA’s recommendations, state of the art precautions),
- Nuclear goods (including approvals, licensing, handling, transport, provisions for import and export, accounting and reporting obligations),
- Nuclear installations (approvals, licensing, construction, operation, decommissioning),
- Radioactive waste (principles of management, responsibilities, national waste agency),
- Supervision (regulatory body, other authorities, technical support, safeguards, proceedings),

- Ensuring and financing decommissioning and disposal of nuclear installations,
- Legal dispositions (liabilities, taxes, financial support, damages, penal provisions).

The nuclear law shall then be implemented via a more detailed nuclear regulation.

3 Additional volume of HLW, LLW and ILW, and the variety of isotopes expected from nuclear power facilities (Small Modular reactors – SMR)

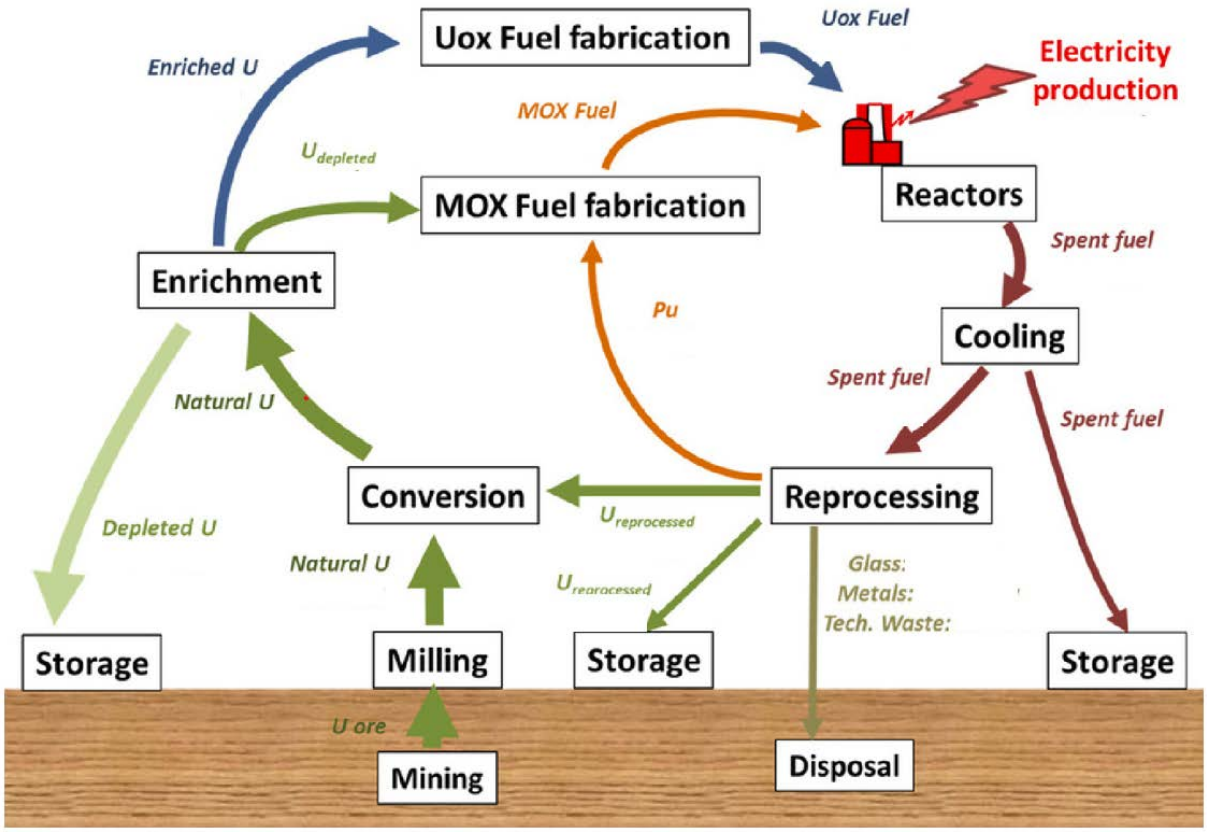
3.1 The nuclear fuel cycle

The raw material for today's nuclear fuel is uranium. It must be processed through a series of steps to produce an efficient fuel for generating electricity. Used fuel also needs to be taken care of for reuse and/or disposal.

The nuclear fuel cycle includes the 'frontend', i.e., preparation of the fuel, the 'service period' in which fuel is used during reactor operation to generate electricity, and the 'backend', i.e., the safe management of spent nuclear fuel, potentially including reprocessing and reuse, and disposal.

The present report will focus on the fuel cycle backend.

If spent fuel is not reprocessed, the fuel cycle is referred to as an 'open' or 'once-through' fuel cycle; if spent fuel is reprocessed, and partly reused, it is referred to as a 'closed' nuclear fuel cycle.



Open and closed fuel cycle advantages and disadvantages as well as examples are given throughout paragraphs 4, 5 and 6. At the beginning a special paragraph summarises the key take-aways of backend HLW waste streams for all available strategic LWR nuclear fuel cycle options.

3.2 Radioactive waste management

Radioactive waste management refers to the safe treatment, storage and disposal of liquid, solid and gas discharge from nuclear industry operations with the goal of protecting people and the environment.

Radioactive waste of various types results from any activity that makes use of nuclear materials, including medical and industrial uses. However, nuclear energy is the most important source of such wastes because of the larger volumes generated and its long-lived nature. Whatever their origin, radioactive wastes must be managed safely and economically.

In general, radioactive waste is separated into three main categories: low-level waste (LLW), intermediate- level waste (ILW) and high-level waste (HLW), depending on its level of radioactivity and the length of time it remains hazardous. Disposal of LLW and most ILW is a mature practice, while most HLW is safely stored in dedicated facilities. The permanent disposal of HLW in deep geological repositories is accepted to be practicable by the scientific and technical community but has yet to be accepted by civil society in many countries.

The activities necessary for managing radioactive waste properly can be categorised into the following steps:

- minimising the amounts created,
- conditioning and packaging to permit safe handling and protection during transport,
- interim storage,
- final disposal.

3.3 General radioactive waste classification[12]

Radioactive waste includes any material that is either intrinsically radioactive, or has been contaminated by radioactivity, and that is deemed to have no further use. Government policy dictates whether certain materials – such as used nuclear fuel and plutonium – are categorized as waste.

Waste other than spent nuclear fuel can be divided into different categories based on its level of radioactivity as intermediate level waste (ILW), low-level waste (LLW), very low-level waste (VLLW) and waste streams that can be cleared from regulatory control.

Waste is said to be “long-lived “if it contains a significant quantity of radionuclides with a half-life of over 31 years.

Low-level waste is generated from hospitals and industry, as well as the nuclear fuel cycle. It comprises paper, rags, tools, clothing, filters, etc., which contain small amounts of mostly short-lived radioactivity. To reduce its volume, LLW is often compacted or incinerated before disposal. LLW comprises some 90% of the volume but only 1% of the radioactivity of all radioactive waste.

LLW has a radioactive content not exceeding four giga-becquerels per tonne (GBq/t) of alpha activity or 12 GBq/t beta-gamma activity. LLW does not require shielding during handling and transport and is suitable for disposal in near surface facilities.

[12] See also next paragraph 3.3.1 “Estonian categorization of radioactive waste”.

Intermediate-level waste is more radioactive than LLW. It generates very little heat (<2 kW/m^3) and must not to be taken into account in the design or selection of storage and disposal facilities. Due to its higher levels of radioactivity, ILW requires some shielding.

ILW typically comprises resins, chemical sludges, and metal fuel cladding, as well as contaminated materials from reactor decommissioning. Smaller items and any non-solids may be solidified in concrete or bitumen for disposal. It makes up some 7% of the volume and has 4% of the radioactivity of all radioactive waste.

Together ILW and LLW form the low- and intermediate level waste (LILW) class.

Exempt waste and very low-level waste (VLLW) contain radioactive materials at a level which are not considered harmful to people or the surrounding environment. It consists mainly of demolished material (such as concrete, plaster, bricks, metal, valves, piping, etc.) produced during rehabilitation or dismantling operations on nuclear industrial sites but also from other industries such as food processing, chemical, steel, etc.

VLLW is disposed of with domestic refuse, although countries such as France are currently developing specifically designed VLLW disposal facilities.

High-level waste (HLW) is sufficiently radioactive for its decay heat (>2kW/m^3) to increase its temperature, and the temperature of its surroundings, significantly. As a result, HLW requires cooling and shielding.

HLW arises from the 'burning' of uranium fuel in a nuclear reactor. HLW contains the fission products and transuranic elements generated in the reactor core. HLW accounts for just 3% of the volume, but 95% of the total radioactivity of produced waste.

There are two distinct kinds of HLW:

- Used or spent fuel that has been designated as waste.
- Separated waste from reprocessing of used fuel.

HLW has both long-lived and short-lived components, depending on the length of time it will take for the radioactivity of particular radionuclides to decrease to levels that are considered non-hazardous for people and the surrounding environment. If generally short-lived fission products can be separated from long-lived actinides, this distinction becomes important in management and disposal of HLW.

HLW is the focus of significant attention regarding nuclear power and is managed accordingly.

Share of different classes of radioactive waste in total volumes in storage and disposal, based on the 2016 inventory data

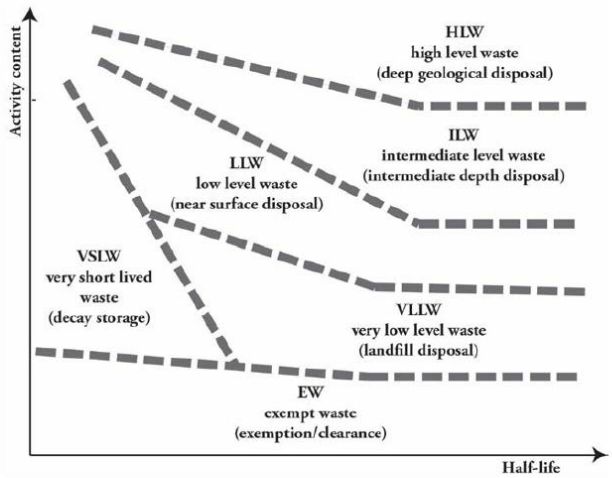
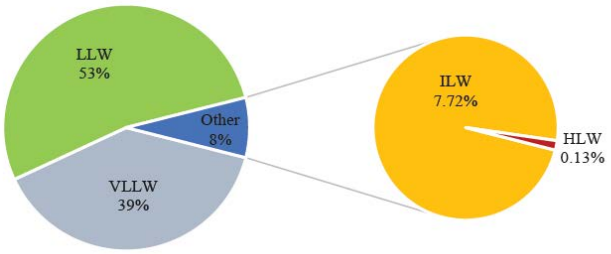


FIG. 2. Conceptual illustration of the waste classification scheme [1].



3.3.1 Estonian categorization of radioactive waste

Radioactive waste[13] is categorized by activity or activity concentration, by half-life, by type of radiation and by heat generation. Categories are established by Regulation No 34 of 4 October 2016 of Minister of the Environment "The Classification of Radioactive Waste, the Requirements for Registration, Management and Transfer of Radioactive Waste and the Acceptance Criteria for Radioactive Waste".

The categorisation of radioactive waste is in accordance with international (IAEA) practice while introducing an additional classification "NORM" – see table below.

Radioactive waste categorization given in the Regulation No 34 of 4 October 2016 is presented in Table below. It describes the types of radioactive waste and of assigned storage facilities.

[13] According to the definition given in the Article 56 of the Radiation Act, radioactive waste is any substances or items which contain or are contaminated with radioactive substances and the activity concentration of which exceeds the clearance levels established on the basis of Article 62 subsection 3 of the Radiation Act and which are not intended to be used in the future.

Type of radioactive waste	Description of radioactive waste	Type of storage facility
Exempt waste	Waste arising from radiation practices, for which the activity and activity concentration or activity concentration on the surface is lower than the clearance levels established under Article 62 (3) of the Radiation Act	Not restricted after release. Handled pursuant to the "Waste Act"
NORM (Naturally Occurring Radioactive Material – substances containing natural radionuclides) waste	Radioactive waste produced as a result of handling raw materials containing substances that contain natural radionuclides (Th-232 and U-238 and radionuclides belonging to their decay series), the specific activity of which is greater than the clearance levels established under Article 62 (3) of the Radiation Act	Storage facility of NORM waste
Short-lived waste	Radioactive waste containing radionuclides with less than a 100-day half-life that will decay below the clearance levels established under Article 62 (3) of the Radiation Act within up to 5 years	Storage room or interim storage facility
Low and intermediate activity short-lived waste	Radioactive waste that contains β- and γ-sources with a half-life less than 30 years and a limited amount of long-lived α- sources (no more than 4,000 Bq/g for one waste package and no more than 400 Bq/g averaged over the total waste package amount)	Interim storage facility or final disposal facility
Low and intermediate activity long-lived waste	Radioactive waste containing radionuclides with half-life higher than 30 years with the activity concentration higher than that for low and intermediate activity short-lived waste and which will generate less than 2 kW/m^3 heat energy by radioactive decay	Interim storage facility or final disposal facility
High level waste	Radioactive waste, which generates more than 2 kW/m^3 heat energy by radioactive decay	Final disposal facility

3.4 Nuclear waste from small modular reactors

Small modular reactors (SMRs, i.e., nuclear reactors that produce <300 MWe each) have expected cost and safety advantages over existing large scale light water reactors (LWRs) and claim inherent safety features and reduced cost.

A renewed interest in small modular reactors can be seen worldwide. The expected advantages of SMRs are in terms of acceptability, financing, and construction time. They also offer other benefits for module manufacturing, simplification of circuits associated with a passive safety approach.

From a reactor physics and fuel management perspective SMRs have low fuel efficiency. The reduced size of the core increases neutron leakage and hence decreases neutron economy.

The low-, intermediate-, and high-level waste stream characterisations [4] prefigure that, "for the same energy supply as large LWRs, SMRs will produce more voluminous and chemically/physically reactive waste than LWRs, which may impact options for the management and disposal of this waste".

Since SMR designs present intrinsically higher neutron leakage most designs are inferior to large scale LWRs with respect to the generation, management, and final disposal of key radionuclides in nuclear waste[14]. Some neutrons that normally sustain the necessary chain reaction are leaked into surrounding structural materials such as steel or concrete, activate them and thus generate additional radioactive waste.

SMRs are also likely to be more uranium intensive for the same energy supply as large LWRs[15]. The intention to reach long cycle lengths influence SMR designers to adopt fuel management strategies resulting in lower fuel discharge burnup[16]. This leads to an increase in natural uranium needs compared to today's LWRs.

Thus, unless compensated for by design changes[17], the final composition of the Spent Nuclear Fuel and associated wastes depend on the initial composition of the fuel, the physical design of the fuel, burnup, and the types of structural materials of the reactor.

Few studies have analysed the management and disposal of SMR nuclear waste streams. There are currently no SMRs in operation[18] and their designs tend to be proprietary. Hence it is difficult to perform in depth SMR waste streams assessments and draw robust conclusions.

Spent SMR fuel characteristics with comparable burnups to large scale LWRs are rather similar. Lower discharge burnups result in lower decay heat and ionization at assembly level.

Light water SMR waste streams will be slightly increased, but their characteristics will be closer to existing large scale LWR waste.

3.4.1 **LWR – High Level Waste**

Two main types of nuclear fuel cycles are currently operated worldwide. In an open or once-through fuel cycle spent fuel is directly disposed after irradiation in a reactor. This is the most common approach.

[14] The probability of neutron leakage is a function of the reactor dimensions and the neutron diffusion length, the latter of which is determined by the neutron scattering properties of the fuel, coolant, moderator, and structural materials in the reactor core. The neutron diffusion length will be the same in reactors that use similar fuel cycles and fuel–coolant–moderator combinations; thus, the neutron leakage probability will be larger for an SMR than for a larger reactor of a similar type. These "lost" neutrons can activate structural materials that surround the fuel assemblies. Each of these physical processes generates radioactive waste. Thus, the final composition of the SNF and associated wastes depend on the initial composition of the fuel, the physical design of the fuel, burnup, and the types of structural materials of the reactor.

[15] Some studies suggest that the natural uranium consumption could double in comparison with an EPR [8]

[16] BWRs have a slightly improved uranium utilization, thanks to their harder neutron spectrum.

[17] E.g.: introducing a neutron reflector to redirect a fraction of leaked neutrons back into the core and/or increase fuel initial fissile loading/enrichment.

[18] Except for the Academik Lomonosova, a non-self-propelled power barge that operates as the first Russian floating nuclear power station.

The closed fuel cycle is a second approach. It aims at recycling spent fuel (or used[19]) already irradiated to benefit from the remaining energy available in nuclear fuel after irradiation. It enables the recovery of plutonium and uranium via reprocessing. Mixed with natural or depleted uranium – a by-product[20] of uranium enrichment – plutonium forms the Mixed-Oxide fuel (MOX).

Spent MOX fuel needs special attention as compared to spent UO_2 fuel.

A closed fuel cycle using used fuel reprocessing and recycling is operated in France, The Netherlands and Japan. In the past it was also used in Germany, Switzerland, Belgium, and others.

3.4.2 **PWR and BWR spent fuel**

Two example cases for light water SMRs fuel types may be regarded as representative:

- BWR - GE Hitachi BWRX-300 and
- PWR – NuScale VOYGR,
  both estimated being commercially available in the early 2030-ies.

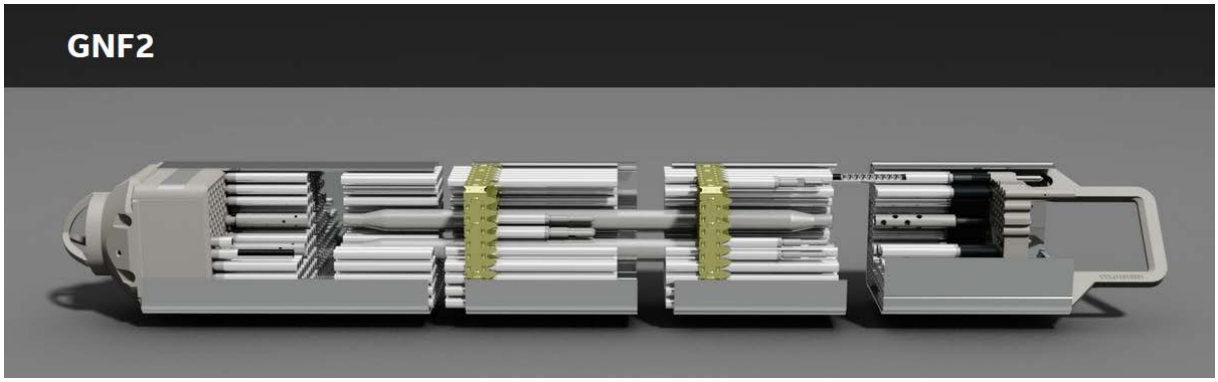
The GE Hitachi BWRX-300 fuel assembly is the standard, full length GNF2 assembly. The 240 fuel assemblies consist each of a fuel bundle and a channel, which surrounds it. Due to their operating conditions BWR fuel assemblies feature several differences as compared to PWR fuels. GNF2 fuel is comprised of 92 fuel rods, including 14 part-length rods, and two large central water rods in a 10x10 lattice array. The part length rods in GNF2 consist of two types of different length: 8 long part length rods and 6 short part length rods. Average fuel enrichment is 3.81 wt% 235U (max. 4.95 wt%). The GNF2 fuel designs foresee axial zoning of uranium enrichment and several gadolinia rods (burnable absorber).

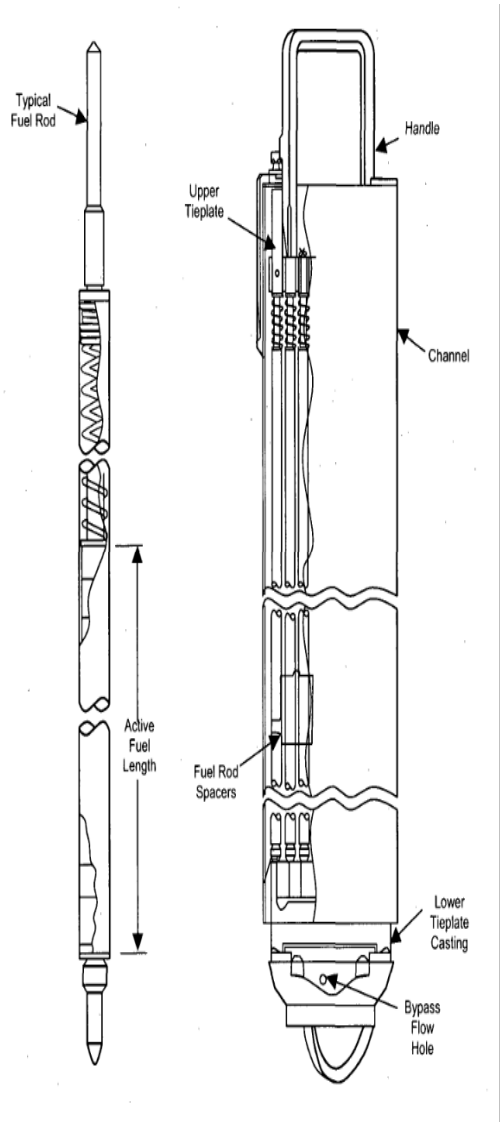
The fuel burned in a BWRX-300 plant is of type GNF2, a next generation development of the currently popular GE14 fuel. The disposal of GNF2 fuel bundles is expected to be very similar to the disposal of GE14 fuel.

BWR fuel assemblies are placed each in a fastened outer channel made of Zircalloy and position the upper end of each fuel assembly in a four-bundle cell. Channels perform several functions like coolant flow path, control rod guidance, structural stiffness etc.

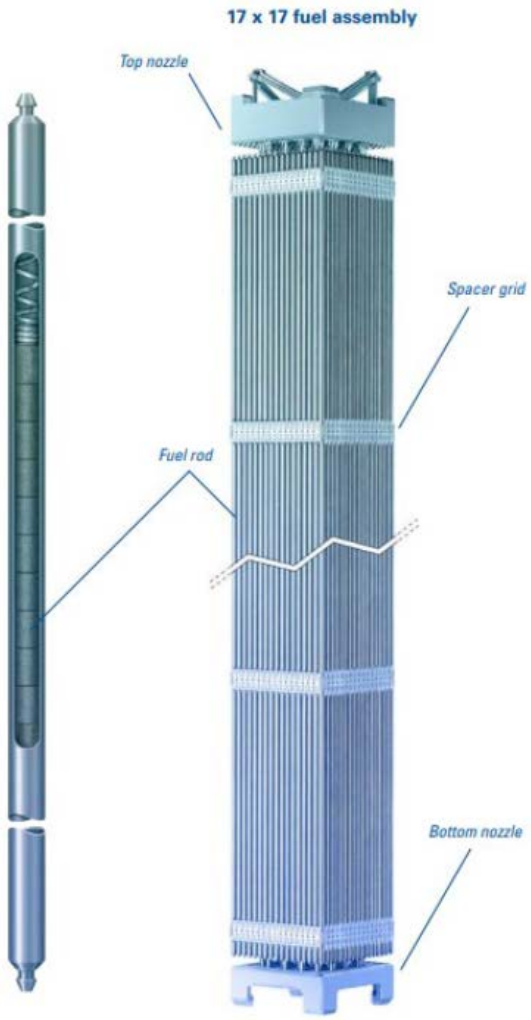
[19] Spent fuel is designated as used fuel when it is considered a resource, not waste.
[20] Tails

Typical GE BWR Fuel Assembly





Typical PWR 17x17 Fuel Assembly (OL3)



The NuScale PWR fuel assembly is shorter in length (~2,4m) with a 17x17 array of fuel rods that has been designed specifically for use with the core configuration of the NuScale reactor. The fuel assembly uses five spacer grids, 24 guide tubes, and a top and bottom nozzle, to provide the structural support for the 264 fuel rods. Each fuel assembly also includes a central instrumentation tube. The reactor core contains 37 fuel assemblies. Average fuel enrichment is max. 4.95 wt% 235U. The fuel is UO2 with Gd2O3 as a burnable absorber homogeneously mixed within the fuel for select rod locations.

There is a vast amount of experience regarding the performance of the fuel, cladding, and structural materials in the fuel assemblies during operations, both wet and dry storage, and final disposal. The fuel assembly radionuclide composition and radiotoxicity are well understood, which will facilitate transport to an interim storage facility and ultimately final disposal.

The fuel assemblies used in the LW-SMRs are based on 17×17 HTP type fuel designs and are essentially identical to the fuel assemblies used in power reactors, except for their length.

Similar experience is available regarding the performance of BWR and PWR fuel, cladding, and structural materials during operations, wet and dry storage, and disposal.

As mentioned before, spent SMR fuel characteristics with comparable burnups to large scale LWRs will also be rather similar.

In Appendix 1 and 2 Swiss KK Beznau (KKB) detailed, representative PWR 14x14 UO_2 and MOX spent fuel characteristics and isotopes are displayed. KKB is the oldest nuclear power plant still in operation today. It consists of two Westinghouse reactor blocks with a net electrical output of 365MW each. They started operations in 1969 and 1972. Both reactors are certified for the use of MOX fuel.

Using MOX fuel increases the natural uranium utilization and hence decreases uranium needs for a given energy generated. However, when irradiating MOX fuel its plutonium fissile vector will be degraded, and more minor actinides will be generated. It further decreases the MOX fuel reactivity and increases challenges for final fuel disposal. Today such MOX fuel is not (yet) reprocessed further.

In Appendix 3 Swiss KK Mühleberg (KKM) detailed, representative BWR 10x10 UO_2 spent fuel characteristics and isotopes are displayed. KKM operated between 1971 and 2019 an early Mark-I GE type-4 BWR with a net electrical output of 355MW.

BWRX-300, is a small modular boiling water reactor. The reactor core contains 240 fuel assemblies. The expected lifetime of the plant is 60 years and during the operation the one reactor will approximately produce 2400 spent fuel bundles (240 + 36 bundles per year, in 24-months cycles) according to GE Hitatchi.

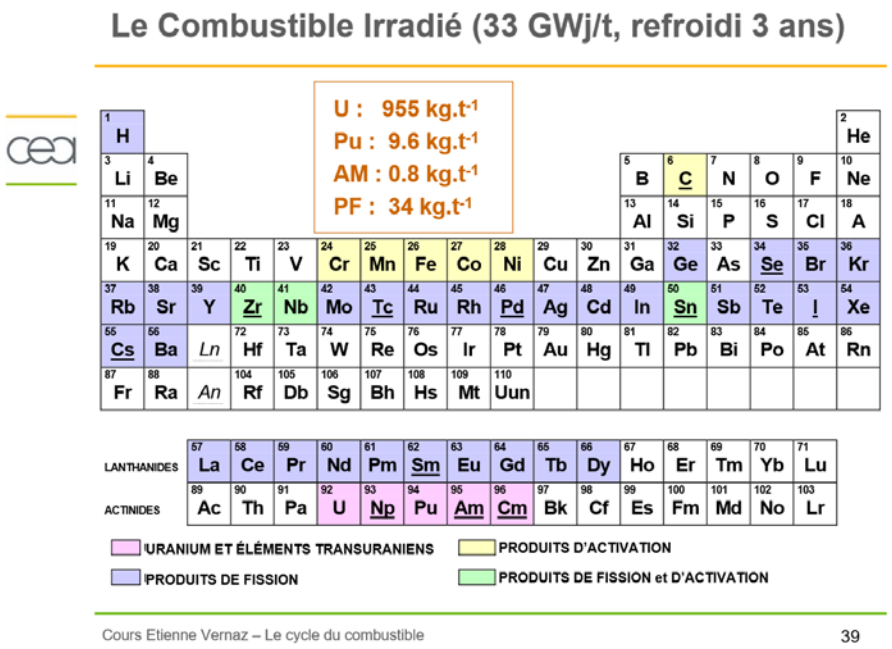
When comparing SNF that will be generated by SMRs with that discharged by LWRs specific SMR operating conditions shall be considered.

Spent fuel assemblies contain a few kilograms of fissile uranium and plutonium. Anti-proliferation safeguards, surveillance and special protection measures are mandatory at all times.

Even if fuel designs will employ UO_2 fuel enriched up to 5 wt % ^{235}U neutron leakage will reduce fuel burnups. SMRs with “once-through” fuel management strategies will attain lower discharge fuel burnups as compared to large reactors[21]. Fuel assemblies are usually expected to be finally discharged after a burnup close to 40 MW_d/kg_{HM}. This is lower than the 60-65 MW_d/kg_{HM} burnup achieved in large PWRs.

[21] Longer fuel cycles, no fuel reshuffling reduce the chance of reaching high burnups.

The typical inventory of a spent fuel assembly (33GWd/t after 3 years cooling time) is given below:



U – Uranium
Pu – Plutonium
Am – Americium
PF – Fission products

Colour code

Uranium and transuranic elements	Activation products
Fission products	Fission and activation products

A representative detailed inventory[22], showing the important actinide elements and the important fission products for a single Yucca Mountain waste package is provided below:

Yucca Mountain Nuclear Waste Inventory per Waste Package by Radionuclide

	Waste Package Inventory (g/pkg)					
Radionuclide	CSNF at 2067	CSNF after 50 Years	DSNF at 2030	DSNF after 87 Years	HLWG at 2030	HLWG after 87 Years
^{227}Ac	2.47E-06	6.27E-06	1.22E-03	1.39E-03	1.91E-04	9.47E-04
^{241}Am	8.18E+03	9.84E+03	2.18E+02	2.15E+02	3.75E+01	3.37E+01
^{243}Am	1.24E+03	1.23E+03	6.73E+00	6.68E+00	5.75E-01	5.70E-01
^{14}C	1.35E+00	1.34E+00	1.81E+00	1.79E+00	0.00E+00	0.00E+00
^{36}Cl	3.23E+00	3.23E+00	4.23E+00	4.23E+00	0.00E+00	0.00E+00
^{245}Cm	1.75E+01	1.74E+01	9.25E-02	9.18E-02	5.43E-02	5.39E-02
^{135}Cs	4.36E+03	4.36E+03	9.74E+01	9.74E+01	1.27E+02	1.27E+02
^{137}Cs	5.90E+03	1.86E+03	9.72E+01	1.31E+01	3.02E+02	4.07E+01
^{129}I	1.73E+03	1.73E+03	3.56E+01	3.56E+01	7.27E+01	7.27E+01
^{237}Np	4.57E+03	5.32E+03	8.14E+01	1.12E+02	9.95E+01	1.04E+02
^{231}Pa	9.17E-03	1.22E-02	2.14E+00	2.14E+00	1.53E+00	1.53E+00
^{238}Pu	1.52E+03	1.02E+03	1.25E+01	6.28E+00	3.91E+01	1.96E+01
^{239}Pu	4.32E+04	4.31E+04	2.21E+03	2.20E+03	5.58E+02	5.57E+02
^{240}Pu	2.05E+04	2.04E+04	4.35E+02	4.31E+02	4.61E+01	4.57E+01
^{241}Pu	2.66E+03	2.40E+02	2.92E+01	4.49E-01	1.22E+00	1.89E-02
^{242}Pu	5.28E+03	5.28E+03	3.02E+01	3.02E+01	3.89E+00	3.89E+00
^{226}Ra	0.00E+00	1.29E-04	4.57E-05	1.80E-04	2.42E-05	2.68E-05
^{228}Ra	0.00E+00	1.90E-11	1.51E-05	8.77E-06	6.00E-06	1.20E-05
^{79}Se	4.19E+01	4.19E+01	6.82E+00	6.82E+00	7.01E+00	7.01E+00
^{126}Sn	4.63E+02	4.63E+02	9.40E+00	9.40E+00	1.70E+01	1.70E+01
^{90}Sr	2.49E+03	7.46E+02	5.22E+01	6.43E+00	1.74E+02	2.14E+01
^{99}Tc	7.55E+03	7.55E+03	1.58E+02	1.58E+02	1.01E+03	1.01E+03
^{229}Th	0.00E+00	2.07E-05	3.24E-01	5.22E-01	3.30E-03	1.05E-02
^{230}Th	1.52E-01	4.32E-01	1.18E-01	2.33E-01	8.12E-04	9.02E-03
^{232}Th	0.00E+00	5.63E-02	2.17E+04	2.17E+04	2.98E+04	2.98E+04
^{232}U	1.02E-02	6.20E-03	1.28E+00	5.39E-01	4.08E-04	1.72E-04
^{233}U	5.76E-02	1.37E-01	5.38E+02	5.38E+02	1.94E+01	1.94E+01
^{234}U	1.75E+03	2.24E+03	4.73E+02	4.79E+02	2.33E+01	4.24E+01
^{235}U	6.26E+04	6.27E+04	2.51E+04	2.51E+04	1.41E+03	1.41E+03
^{236}U	3.84E+04	3.85E+04	1.25E+03	1.25E+03	5.99E+01	6.03E+01
^{238}U	7.82E+06	7.82E+06	6.84E+05	6.84E+05	2.37E+05	2.37E+05

Source: Sandia National Laboratories 2008a; Table 6.3.7-4a

[22] Two different types of waste packages are represented: one package that contains commercial spent fuel inventory, (CSNF) with either 21 PWR assemblies or 44 BWR assemblies, and a co-disposal waste package that combines the defence spent fuel (DSNF) and high level vitrified wase inventory (HLWG). The inventories do not include any Mixed Oxide (MOX) fuel or Lanthanide Borosilicate (LaBS) waste.

3.4.3 **Separated waste from reprocessing**

Following commercial irradiation a spent fuel assembly contains 94 % uranium, 5 % fission products and 1 % plutonium. Both uranium and plutonium isotopes have the potential to be reused for further energy production.

The main current method for the separation of spent fuel is the PUREX process (employed at Orano's La Hague), which, in a series of complex chemical operations, separates the plutonium, uranium and other transuranic elements from the remainder of the spent fuel.

Long-term storage of fission products requires the stabilization of the waste into a form that will neither react nor degrade for extended periods. This is obtained through vitrification at high temperature of fission products which make up around 4% of the spent fuel mass, into a standardized CSD-V package (at La Hague[23]).

It is regularly stated that reprocessing of spent fuel will reduce up to a factor of 5 the volume of final waste.

CSD-V

Package volume: 180 litres
Package mass : 489 kg

The average activity is between $2{,}1.10^{10}$ et $5{,}1.10^{10}$ Bq/g for finished package at production.

Main contributing radionuclides:
α : ^{244}Cm, ^{241}Am
βγ- sl : ^{137}Cs, ^{137m}Ba, ^{90}Sr, ^{90}Y, ^{147}Pm, ^{134}Cs, ^{106}Ru
βγ-ll : no predominant long-lived βγ radioelement

Average thermal power is approximately 2.500 W/package at production. This value is divided by 4 after 50 years of radioactive decay and by 10 after 100 years. The increase of burnup rates induces an increase in thermal power on the production date (around 2,2 kW/package).

Potentially toxic chemical elements per package: Boron: 17kg, Uranium: 2kg, Nickel: 1kg, Chromium: 23 kg, Cadmium: 190g, Selenium: 99g, Antimony: 25g

In Appendix 4 characteristics and isotopes of a representative "CSD-V" universal canister containing fission products (residues) from Swiss reprocessing campaigns are displayed. The HLW residues are contained in a boron-silicate glass matrix and placed in a steel container.

3.4.4 Operational L&ILW and decommissioning waste

During the operation of a reactor, different types of low and intermediate radioactive waste are generated. This waste includes filters used in water and air treatment, worn out components and industrial waste that has become contaminated with radioactive substances. This waste must be conditioned, packaged and stored prior to its disposal. Most of this waste (by volume) has low

[23] Other reprocessors, such as Sellafield or Mayak produce waste packages different in size, quality and content.

levels of radioactivity (VLLW or LLW). It includes decontamination solutions and materials, contaminated building materials, pieces of metal or wood, electric wires, etc.

In principle, it can be assumed that the ILW generated in an LWR-type SMR consists mainly of ion exchange resins used for treatment of process waters (reactor coolant), as it is in a typical NPP.

For an LWR-type SMR unit, decommissioning waste includes potentially all the integrated reactor components (reflector, etc.) and the containment vessel itself that are all likely to fall in the ILW or LLW category.

The decommissioning of full factory-assembled reactors (i.e., SMRs) may be technically less demanding as they could be transported back to the factory in an assembled form.

VLLW and LLW typically account for more than 95% of the volume but less than 2% of the radioactivity of all radioactive waste. It does not generally require significant shielding during handling, conditioning, and interim storage. The waste is suitable for disposal in engineered near surface facilities.

ILW generally contains significant amounts of long-lived radionuclides and therefore requires ILW requires shielding during handling, conditioning, transport, and interim storage. It also requires disposal at depths that provide isolation from the biosphere over the long term.

In case that used fuel reprocessing is envisaged in France all separated structural metallic spent fuel components will be compacted into standardized CSD-C packages (at La Hague). Such metallic residues are categorized intermediate-level long-lived waste.

Package volume: 180 litres
Package mass: 700 kg
Medium mass of compacted metal: 600 kg

Medium activity: 4,1.108 Bq/g per package
Main radionuclides:
α : no prominent α radionuclide
βγ-sl : ^{55}Fe, ^{60}Co, ^{3}H, ^{137}Cs, ^{90}Sr, ^{90}Y, ^{137m}Ba
βγ-ll : ^{63}Ni

Thermal power: 12W/package after 25 years

Toxic elements per package: Chrome: 36 kg (mainly from the stainless steel of the end pieces), Nickel: 27 kg (mainly from structural elements in nickel alloy and the stainless steel of the end pieces), Lead: 1 kg, Uranium: 730 g, traces of Antimony and Selenium

At the end of its operating life, a reactor is shut down and eventually dismantled. During dismantling, contaminated and activated components are separated, treated and if necessary managed as radioactive waste. The largest volumes of radioactive waste generated are in the

VLLW or LLW classes. Smaller volumes of ILW are also generated. Most of the waste (by volume) from dismantling is, however, not radioactive and can be handled as industrial waste, in accordance with the country's regulations.

Reactor dismantling is divided into several areas - here is an overview of the key data:

- Essentially it includes the dismantling of plant components in the machine building, such as turbines and generator, condenser and water separator reheater, pumps and fittings.
- It includes further the dismantling of plant components in the reactor building, such as steam generators[24], reactor coolant pumps and reactor coolant lines.
- Further it includes the dismantling of the "plant core", i.e., the dismantling of the upper and lower core structure, the reactor pressure vessel, and the so-called biological shield.
- And it includes the dismantling of remaining systems and plant components. These are, for example, parts of the ventilation systems, freight elevators, the crane system in the reactor building and parts of a large material lock.

The afferent waste volumes are difficult to estimate and depend on many boundary conditions such as initial fuel enrichment, reactor operation or the national regulation and competent authority (e.g., allowing or not a release threshold). A typical 1000 MW(e) PWR or BWR produces between 5.000 and 10.000m^3 of decommissioning waste.

As example of an order of magnitude a German 1.400MW PWR is expected "produce" roughly 6.000 – 7.000m^3 of LLW and ILW (10-15%, resins, plant core components etc.). For a same power BWR the waste volume is expected to increase with ~25%.

As preliminary conclusion it can be stated that at comparable power production, back-end inventories and environmental footprint for SMRs will be greater as compared to large scale PWRs or BWRs.

3.4.5 Advanced modular reactors (AMR) using novel and innovative fuels, coolants[25]

3.4.5.1 Molten-salt- and sodium-cooled AMRs

Waste streams expected from first SMRs (light water PWR and BWR type) will be different from those of advanced modular reactors (AMR) using novel and innovative fuels, and coolants.

For example, molten-salt- and sodium-cooled AMRs employ highly corrosive and pyrophoric fuels, liquid metal or salt coolants and graphite moderators and reflectors that, following irradiation, will become highly radioactive.

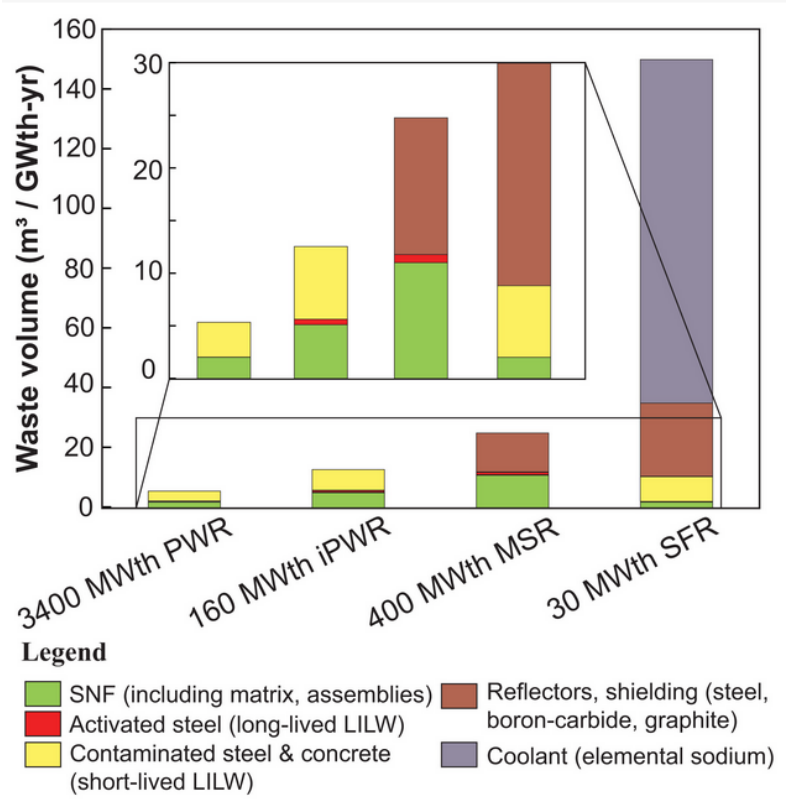
Primary coolant (e.g., in the form of molten salt, or sodium) will be circulated and heated in the reactor core. It comes in direct contact with the active core, reactor components and heat exchange systems.

In the case of sodium- and molten salt–cooled AMRs, the primary coolant will be chemically reactive, heated to temperatures >500 °C, and highly radioactive. Under these extreme conditions, reactor components can have a shorter lifetime than the standard PWR (60 years), and this will increase decommissioning LILW volumes. In addition, AMRs will introduce uncommon types of LILW in the form of neutron reflectors and chemically reactive coolant or moderator materials.

[24] Only for PWRs
[25] See [4],]13]

Molten salt– and sodium-cooled AMRs will also experience enhanced neutron leakage. AMR designs may seek to offset the leakage by using neutron reflectors and/or fuel enriched to >5 wt % initial fissile concentration, fuel burnups will be lower than for larger molten salt– and sodium-cooled reactors.

The energy-equivalent mass of SNF generated by SMRs and AMRs can be estimated considering the respective burnup[26].



Energy-equivalent waste volumes, by waste type, for various SMR designs—including the NuScale iPWR, the Terrestrial Energy IMSR, and the sodium-cooled Toshiba 4S SMRs.

In general, long-lived LILW consists of near-core reactor components that have become radioactive or activated after absorbing neutrons leaked from the core. This activated steel contains radioisotopes with half-lives longer than several thousand years (e.g., ^{59}Ni, ^{14}C, ^{94}Nb, ^{99}Tc, ^{93}Zr, ^{93}Mo, and ^{36}Cl) and so, should be disposed of in a geologic repository.

*3.4.5.2 **High temperature gas cooled (HTG) – AMRs***

HTG-AMRs use gas as coolant at high operating temperatures, and graphite[27] moderators (for which no disposal routes are yet established), but otherwise have unique design characteristics.

[26] Whereas a PWR with a burnup of 55 MW_d/kg discharges ~6.5 t SNF/GW_{th}-y, a non-water-cooled SMR may discharge 1.5 to >36 t SNF/GW_{th}-y. These figures, however, solely reflect the mass of uranium, actinides, and fission products in the SNF and neglect contributions from salt or sodium constituents in or around the fuel matrix. Such low-density materials contribute little to mass-based SNF estimates but nevertheless, will contribute to volume-based estimates. [4]

[27] Graphite as a moderator generates relatively large amount of irradiated graphite waste during operation and decommissioning.

Their thermal capacities range from 100 MW to more than 600 MW, while the electrical capacities range from 35 MW to 300 MW.

In a 200-MWe NPP-sized pebble bed[28] reactor, the estimated amount of irradiated graphite is up to 17 tons annually. Irradiated graphite waste typically contains significant amounts ^{14}C (half-life 5730 years) and ^{36}Cl (half-life~300 000 years) and small amounts of corrosion and activation products, some fission products and trans-uranic impurities.

Most of the graphite waste falls within the LILW category. However, it should be also noted that some of the fuel-contaminated graphite may be classified as HLW.

Handling, storage and disposal of graphite waste can be hazardous and requires specific attention.

Nuclear waste streams from HTG-AMRs include also liquid waste, off-gases and irradiated components, such as the reactor vessel.

[28] The "pebble bed" design uses spherical fuel elements (graphite pebbles), typically tri-structural isotropic (TRISO) particles. In most designs the fuel particles consist of UO_2, but some designs can use PuO_2 and MOX as fuel. The fuel enrichment varies between 8 -20 %. Separation of fuel kernels (TRISO particles) from the graphite pebbles might be needed to be prior to disposal, e.g., by low or high temperature acid treatments. The fuel can be further reprocessed, stored, and possibly disposed in a similar manner as spent fuel from LWRs.

4 Technological options and research on the ultimate disposal of spent fuel and HLW from reprocessing

Studies[29] imply that the nuclear waste management metrics (including both the mass of spent nuclear fuel and LILW volumes) are negatively impacted in the SMR design compared to the large LWR, i.e., more spent fuel and low-level waste are produced on an energy equivalent basis in the SMR versus the large LWR.

Differences in fuel cycle performance will have an impact on nuclear waste management, resource utilization and economic and financial concerns. The smaller axial and radial core will yield increased neutron leakage versus a large core, resulting in lower fuel utilization.

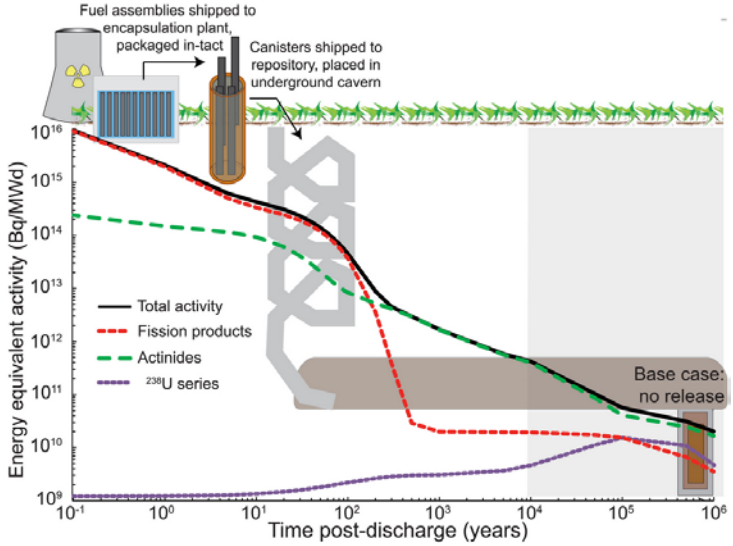
Additionally, because the maximum enrichment just under 5 wt % SMR designs may envisage varying enrichments to control power peaking, thus the average core enrichment may be less than in a large LWR and will yield lower fuel discharge burnups.

Fuel assembly designs used in the light water SMRs are based on current large scale PWR or BWR fuel designs, except for their smaller height[30]. As the fuel and reactor types are like those used in the current LWRs, the SMR spent fuel features with comparable burnup will be rather similar.

Such SMR spent fuels could be managed with the same or similar equipment as current LWR spent fuels, e.g., in pools and hot cells, or for packing, transport, storage, reprocessing, final repository[31].

The decay heating power is one of the most limiting factors when the packing density of the final repository is defined. Several decades of prior interim cooling time for HLW will be necessary.

The duration of interim storage is factored into repository dimension calculations because the SNF decay power decreases over time – see below post discharge time vs. energy equivalent activity.



[29] See [4], [13]

[30] Not the case of BWRX-300.

[31] The generally lower discharge burnup characteristics for the small reactors facilitate the final disposal safety case at the assembly-level in terms of lower radioactivity and the consequent lower heating power. Also, the concentration of mobile nuclides in the spent fuel is smaller. Concerning spontaneous fission rate the main driver in spent fuel is the discharge burnup.

Used nuclear fuel may be treated as a resource or simply as waste. This paragraph will deal with fuel cycle strategies, fuel management, interim storage, and final repository options.

4.1 Nuclear fuel cycle

Spent fuel is generated from the operation of nuclear reactors of all types, including research, isotope production, power production, district heating and propulsion reactors. By volume, HLW forms less than 1% of the global volume of radioactive waste, but it consists of about 95% of the total activity of the radioactive waste. The activity level of HLW is high enough that heat generation must be considered in the design of the waste management facilities. In countries where spent fuel is classified as waste, it is classified as HLW.

4.1.1 The 'open cycle' and 'closed cycle'

Management of spent fuel differs from one country to another. The currently envisaged strategies ensure a safe and cost-effective overall management of spent fuel and can be described as follows:

- The 'open cycle', 'once through' or 'direct disposal' strategy, in which spent fuel is considered as waste,
- The 'closed cycle' (including the 'partially closed cycle') strategy, in which spent (or used) fuel is considered to provide a potential future energy resource.

4.1.1.1 Open cycle: facts, practices and options

In the open cycle option, spent fuel is interim stored for several decades to allow the decay heat to be reduced before geological disposal. Interim storage starts off in the reactor pool after spent fuel unloading. To avoid reactor pool saturation spent fuel may be later-on transferred to centralised pools for continued wet storage. This is the case among other in Sweden, Slovakia, Bulgaria.

Central Interim Storage Facility (CLAB), Sweden. Image: SKB

Long-term pool storage leaves the spent fuel always accessible for inspection and repair, which is a technical advantage. If required, before final repository suited (state of the art) transport casks will be provided that will respond to all meanwhile evolved technical and licensing requirements. The spent fuel may be packed under water into final suited DGR containers and moved with an onsite

shuttle cask (if the centralised pool is on the same premises with the DGR facility) or transported to the final repository site (where suitable hot cells type facilities for handling and packing shall be available). On the other hand, in terms of capex and opex pool storage is more expensive than dry storage. After the Fukushima (pool) accident safety authorities are rather restrictive.

Optionally, spent fuel may be transferred from the reactor pool to a dry interim storage facility. Different types of such facilities can be considered:

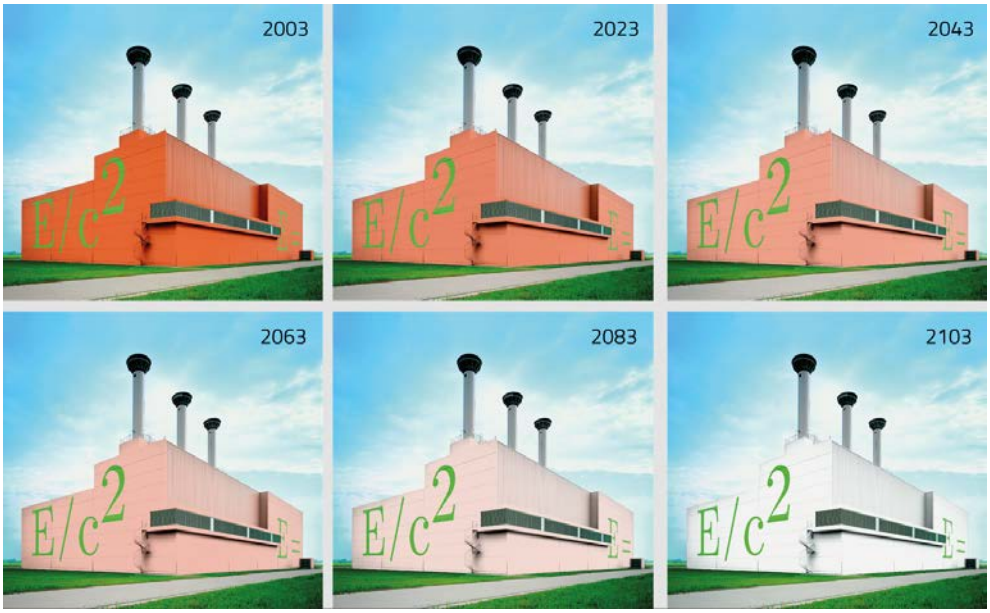
- On-site or centralised off-site dry storage in dual purpose casks (i.e., for transport and storage) in protected buildings, as is the case among other in Switzerland and Germany. Typically, a dual-purpose cask contains a few dozen spent fuel assemblies (up to 89), the amount depending among other on the fuel type (more BWR or less PWR), their thermal power (MOX and high-burnup fuels are “hotter”) and dose rates. In some situation the cask size (and capacity, number of contained fuel assemblies) may be limited by roads or bridges to be passed or access to certain facilities.

Dual-purpose casks in dry storage hall at Dukovany NPP (Image Dukovany)

- On-site or centralised off-site dry storage in metallic canisters placed in concrete structures (horizontal or vertical concrete modules), under the open sky or in buildings. In such case the spent fuel storage function is separated from the transport function and a special and fully licensed transport vehicle/shuttle must be always provided. Such open sky storage systems are common in the USA. Slovenia also made this choice but places the storage systems in a building. Canister capacities are in the same range as for dual purpose casks.

Holtec's HI-STORM 100 free-standing above-ground system (Image Holtec)

- Centralised dry storage in shaft (vault or silo) type structures/buildings, as is the case in The Netherlands (for HLW from reprocessing and spent research reactor fuel).



Covra's HABOG facility -outside view in time (Image Covra)

Every 20 years the HABOG building is repaint in lighter colour to reflect to the outside the nuclear decay happening inside.

Covra's HABOG facility for HLW -inside view (Image Covra)

The Netherlands made the choice to interim store HLW from reprocessing and research reactors for 100 years before final disposal. This strategy will allow the HLW to decay and sufficiently cool down, and it allows Covra to develop a final disposal solution (national or regional), meanwhile managing and developing the necessary financial resources. A hot cell

for HLW handling and packing/unpacking is available. Before final repository suited transport casks shall also be available.

The above HLW dry interim storage options have all their advantages and some downsides. Here are some examples:

- Loaded dual purpose casks can be transported at all times - as long as transport licenses are valid and not expired. Before final repository spent fuel should be repacked into final disposal suited containers. Slowly the industry understands better the very long-term spent fuel ageing process inside sealed dry interim storage casks.
  Opening of such casks and spent fuel repacking must be conducted under water, i.e., in a pool or a large hot cell. In a few decades on-site reactor pools may be already dismantled and the dry storage casks' transport license expired, thus making spent fuel evacuation problematic. Early, periodic evacuation should be envisaged.
  Transport means to final disposal sites and there, large hot cells for spent fuel handling and repacking shall then be provided, as well as the DGR-suited storage casks Damaged fuel needs special attention, must be managed differently. An unpredictable amount of financial risk is associated.
- Dry storage systems in metallic canisters placed in concrete structures are a sound technical option and financially more convenient. As said earlier, the spent fuel storage function is separated from the transport function and a special, state-of-the-art transport vehicle shall be always available. The transport shuttle will be continuously maintained up to date from a technical and licensing point of view over the entire interim storage time. However, it will be able to evacuate stored spent canisters only in small batches, not all together at any time as for dual-purpose casks. Here too, the spent fuel should be repacked before final repository, which means the canisters will be opened and emptied. Under water or hot cell handling will be required. Alternatively, the canisters might be placed in an overpack. Damaged fuel also needs special attention, must be managed differently. In all cases an important financial risk is associated.
- Compact and centralised dry storage in shaft type structures in buildings are best suited for HLW vitrified canisters stemming from reprocessing. It can also be provided for spent fuel assemblies. HLW canisters are stacked on top of each other in vertical shafts, shielded and cooled via natural convection. Because of their length stacking is unlikely for full size fuel assemblies, although for short SMR fuel this might be easier. Handling and transport to the final disposal site in a far future must be provided unless the storage facility is on the same premises with the disposal facility where an onsite shuttle cask might be sufficient. For conditioned HLW canisters the associated financial risk of this solution is lower than for spent fuel interim storage solutions because all technical spent fuel related uncertainties have been eliminated by reprocessing. On the other hand, reprocessing of spent fuel requires earlier important expenses that could be offset by customised financial management.

After a period of interim storage, the spent fuel will be encapsulated in a robust, corrosion resistant container to meet disposal acceptance criteria and will be disposed of in a deep geological repository.

In some cases, the long-term interim storage task lies within the responsibility of an operator and in other cases the spent fuel is taken over by national state agencies at the time or shortly after it is evacuated from the reactor pool. Respectively different and suitable management and financing models will be required.

For most operators and/or national waste agencies the choice of an interim storage solution will depend on the capex, opex, financial risk, public acceptance, safety, and licensing requirements.

4.1.1.2 Closed cycle: facts, practices and options

In the closed cycle the spent fuel is reprocessed in order to recover valuable fissile materials (uranium and plutonium).

In reprocessing spent fuel is separated into several main components: uranium, plutonium and HLW (containing minor actinides, fission and activation products).

HLW (along with other waste such as LLW and ILW) resulting from reprocessing is then stored to allow the decay heat to be reduced pending future disposal, normally in a DGR – see above.

Uranium and plutonium can be recycled as nuclear fuel for reactors.

Over the second half of last century, several factors have led to question of pursuing the once-through strategy as the sole strategy for spent fuel management. In particular:

- the cost estimates for direct disposal significantly increased,
- the increase in nuclear energy use required consideration of additional fissile quantities,
- the commercial constraints for recycling have changed, driven by higher uranium prices and improved understanding of the long-term behaviour of the "by-products" of reprocessing, which enables a significant optimization of the repository volume,
- recycling has demonstrated its technological and commercial maturity.

The separated plutonium can be recycled into MOX fuel (together with depleted uranium stemming from uranium enrichment). MOX fuel has been used for decades in LWRs worldwide and in a few fast (generation IV) reactors[32], where the energy value of the uranium and plutonium can be better utilized[33].

The main benefits of reprocessing and recycling are:

- for the frontend of the fuel cycle, it increases energy security by potentially saving up to 30% natural uranium,
- for the backend, it facilitates disposal by reducing volume by a factor of 5 and toxicity by a factor of 10. In addition, disposal is optimized with standardized types of waste[34].

Conditioned vitrified HLW contains the fission products but no fissile material. The reprocessing strategy may offer several advantages[35]:

- No spent fuel ageing uncertainties after long time interim dry storage,
- No future spent fuel conditioning before final disposal,
- Due to absence of fissile materials in HLW potentially no safeguarding requirements.

[32] France in the past, and today in the Russian Federation.
[33] There is almost no other use for ^{238}U other than in fast breeder reactors.
[34] HLW vitrified canisters contain no fissile materials and potentially do not require NPT safeguards (to be confirmed).
[35] Also with financial impact.

HLW resulting from reprocessing must also be stored to allow heat decay before final disposal[36]. It is imaginable that the reprocessor provides interim storage and the HLW from reprocessing is returned only later, possibly just before final disposal.

'Fully closed recycling' requires as a prerequisite spent fuel reprocessing and MOX multi-recycling in thermal reactors and/or Pu-breeding and MOX recycling fast reactors[37]. It also requires recycling of reprocessed uranium (RepU) which can serve for Pu-breeding in fast reactors.

So far, no country has completely closed the fuel cycle on an industrial scale[38].

'Partially closed cycle' can have different significations. Today in most cases it means reprocessing of spent fuel and "thermal" MOX[39] fuel fabrication followed by MOX recycling[40] in LWR power reactors.

Another 'partially closed cycle' possibility consists in only RepU recycling via ERU fuel elements burned also in LWRs[41].

Reprocessing and thermal MOX fuel production capacities currently only exist in France in Orano's La Hague (1.700 t_{HM}/y) and Melox plants (theoretically 195 t/y)[42]. Spent MOX fuel reprocessing is demonstrated but not yet a standard routine[43].

4.1.1.3 Plutonium recycling

For some plutonium and reprocessed uranium are reusable, valuable materials. Others see Pu-separation as a proliferation risk, a "devil's tool". Others again consider it a liability, as mentioned above.

Some background history: about 30-40 years ago Pu was largely considered a recyclable material to be used in fast breeders. Thermal Pu-recycling was only seen as a temporary solution to avoid storage costs. Unfortunately, most fast breeder programs failed, except in Russia, and Pu was no longer considered a valuable resource, but a liability that causes additional costs.

This situation is changing today, and in recent years many countries consider generation-4 advanced (also modular) reactors which implicitly brings reprocessing into the discussion again.

If a country like Estonia thinks about reprocessing and recycling in order to act sustainably, save resources, recover valuable materials, prior careful consideration is recommended.

Plutonium is formed in nuclear power reactors from ^{238}U by neutron capture. Plutonium is radioactive (most isotopes are alpha emitters), toxic, radiotoxic and its uneven isotopes can fission when a neutron hits the nucleus. Alpha-particles can be easily shielded (e.g., already by a sheet of paper) but, because of their radiotoxicity, alpha-emitters shall no case be incorporated by humans.

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[36] Normally also in a DGR or, if demonstrated feasible in DBDs

[37] Through the combination of fast breeder reactors, reprocessing and LWRs, the earth's uranium stock could provide about 30 times more energy than if only the ^{235}U were burned.

[38] Russia's Rosatom communication suggest an advanced status.

[39] "Thermal" MOX fuel is designated for current LWRs (not for fast breeder reactors). "Thermal" because fission is produced by moderated, slow so called "thermal" neutrons.

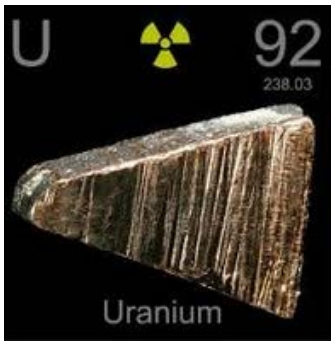
[40] In France, The Netherlands, Japan

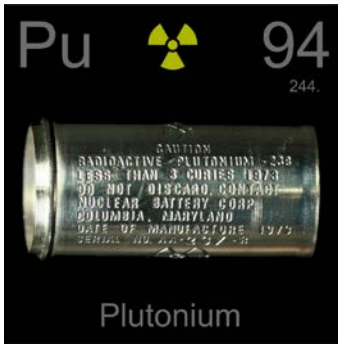
[41] Recycling RepU from reprocessing of LWR fuel in a pressurized heavy-water reactor, such as a CANDU reactors, is developed in China. In the past RepU was sometimes used to "perfume" enriched natural UF_6 within allowed limits which also can be seen as recycling.

[42] The MOX plants in Sellafiedl (MDF and SMP), Hanau/Germany, Dessel/Belgium and Cadarache/France have now all been shut down after successful operation.

[43] Spent MOX fuel is ideally suited for recycling in generation-4 reactors.

Some Pu-isotopes decay differently. ^{241}Pu, a beta-emitter, will form americium that has important consequences during storing, handling and further processing. In general Pu feels warm, where other metals feel colder. During its radioactive decay energy is released in form of heat. ^{238}Pu is a good example finding application as nuclear spacecraft batteries or long-lived pacemakers. But the thermal power of Pu imposes cooling and some restrictions in handling and processing.





MOX powder

Pure Pu is a chemically complex element. Plutonium oxide has completely different properties than those of pure metals. It turns out to be safer[44] and is used in civil applications.

Plutonium presents a proliferation risk and strict safeguard measures are imposed during all handling and processing. It shall be noted that (civil[45]) Pu separated from power reactor operation (including light water PWRs and BWRs) is not weapon-grade material. In most cases[46], the higher the fuel burnup[47] the less suitable it will be.

Opting for a closed fuel cycle strategy will allow the reuse of uranium (as reprocessed uranium) and plutonium that otherwise would be disposed of together with the spent fuel[48]. Both can be recycled as fresh fuel, saving up to 30% of the natural uranium otherwise required.

Opting for a closed cycle strategy will also imply a different way of managing the HLW regarding storage and disposal compared to the open cycle approach. Another positive aspect is that the HLW vitrified packages, in contrast to spent fuel, do not fall under IAEA safeguards, and present no proliferation risk[49].

The time span after reactor unloading of spent fuel until its reuse as MOX fuel will take 15 years or more. Eventually when the reactor is shut down the remainder of spent fuel may be reprocessed (and benefit of HLW conditioning) but not recycled unless another fleet reactor or third-party reactor takes over the recycling task.

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[44] PuO_2 is a stable ceramic material, warm at touch, barely soluble in water, with a high melting point (> 2.300°C). Studies show that it is not transported through ground water.

[45] Reactor-grade plutonium is defined as that with 19% or more of ^{240}Pu (which has a relatively high rate of spontaneous fission with consequent neutron emissions, making it unsuitable for military purposes).

[46] There are reactors available for electricity production that could be used to produce weapon-grade Pu. Their particularity is that used fuel can be extracted at any time, i.e., when its fissile Pu-content is best suited for the military task (i.e., after a few months of irradiation).

[47] British Magnox reactors used for production of military plutonium in their early years (to 1964) were run at about only 0,4 GWd/t burn-up.

[48] About 96% is uranium, of which less than 1% is the fissile ^{235}U (often 0.4-0.8%) and up to 1% is plutonium.

[49] However, the fact that current reprocessing technology involves separation of weapons usable plutonium has led to concerns about the spread of the technology to many countries.

While the HLW storage and disposal options will reduce risks and simplify the waste management exercise the reuse of reprocessed uranium and plutonium is a more complex task.

A specially adapted and licensed reactor is needed to burn Pu-containing MOX fuel. Ideally this would be the "reactor of origin", whose spent fuel was reprocessed. Some minor adaptations are required for MOX use:

- Reactivity control devices
    - due to higher energy neutron spectrum (higher Pu content): additional RCCAs, increased boron concentration
- Fuel building adaptation
    - reinforcement of the crane (hardware and software)
    - fresh MOX reception and storage under water
    - reinforced plant safeguards (cameras in fuel building, …)
- Fresh MOX fuel transport in special casks
    - to reduce the risk of excessive exposure of the operators during handling and
    - to improve transport safety and nuclear materials safeguards
- Operator training (fuel handling, core monitoring)

During reactor operation with MOX fuel there is no significant impact on plant availability. As well, operational flexibility will meet the electrical grid demand. There is no increase of waste release to the environment, no impact on plant workers integrated dose. In case of supply chain disruptions MOX fuel can be replaced by uranium fuel.

However, at reactor shutdown spent fuel would be left over, whose plutonium cannot be burned anymore in that reactor[50]. Another newer fleet reactor or a third-party reactor could take over the Pu-burning. Otherwise, separated plutonium becomes a liability and the owner must eventually take care of it[51].

Today no light water SMR provider proposes usability of MOX fuel. LWRs can be also later-on adapted and licensed for MOX use, which is in fact the case today of all MOX-using LWRs.

MOX fuel assemblies are structurally identical to the uranium fuel assemblies used in the respective LWRs. They have a comparable burnup potential and are designed with a comparable average reactivity. While the fuel rods in PWR uranium fuel assemblies usually have all the same enrichment, fuel rods with different enrichment levels and additional water rods are used in BWR MOX fuel[52]. Regarding reactor core management, there are no fundamental restrictions. Up to 40% of a LWR reactor core may be loaded with MOX fuel.

Since the MOX production costs are an order of magnitude higher, thermal MOX fuel assemblies are not necessarily designed in a neutronically optimized manner, but rather to accommodate as much Pu as possible, viewed as a liability. In France the initial carrier material natural uranium is replaced by depleted uranium[53].

[50] In addition, MOX use, needing longer cooling times, stops several years before reactor shutdown. The entire fuel of the reactor core will be left over together with the spent fuel remainder of the reactor pool.
[51] Or even be declared waste, depending on the state's legislation. In French law, for example, it stays a reusable material as long as reuse can be envisaged.
[52] To flatten the power density. The reduced thermal flow in MOX fuel influences the control element effectiveness. The void coefficient becomes more negative, and the boron efficiency decreases.
[53] For the same reason, stemming from enrichment tails.

Due to the gamma and neutron radiation, handling of fresh MOX fuel assemblies is more difficult. For reasons of radiation shielding, when delivered at the power plant they are usually not taken to dry storage, but straight to wet storage in the reactor pool[54].

Spent MOX fuel assemblies must be cooled for a significantly longer period before they can be transferred to dry interim storage casks or taken to final storage[55]. Therefore, MOX use usually stops several years before reactor shutdown to permit speedy pool emptying at decommissioning.

For reasons of cost, MOX is usually used in PWRs, since BWR MOX fuel assemblies lead to higher manufacturing costs given the complexity of the BWR fuel design.

The cost-effectiveness of using MOX depends primarily on the raw material prices. Natural uranium prices have been at a low for a long time. Under such market conditions, MOX fuel was significantly more expensive than uranium fuel. However, there are several reasons why nuclear power plant operators use MOX despite adverse market conditions:

- the fuel price in a nuclear power plant has little impact on the electricity production costs,
- reprocessing and recycling strategies were based on long-term contracts or on national legal frameworks, and
- separated Pu by reprocessing may represent a liability that the nuclear power plant operators would like to reduce, especially in the event of a foreseeable end to the power plant's service life.

It shall be also noted that today only France is proposing industrial reprocessing and MOX fuel fabrication and a large part of its capacities are dedicated to the French nuclear cycle. France provided itself with almost the entire necessary industrial infrastructure and operating capabilities. Approx. 10% the electricity production stems from the deployment of thermal MOX fuel.

MOX was and is used in Europe and in Japan. About 40 reactors in Europe (Belgium, Switzerland, Germany, The Netherlands, and France) are or were licensed to use MOX. Today in Europe MOX fuel is used in France and The Netherlands. Another three MOX-licensed reactors are still operating in Switzerland, out of which only one could theoretically use MOX again. The 24 (out of 56) French reactors' MOX burning capacity is fully dedicated to the French nuclear cycle. In Japan about ten reactors are licensed to use MOX and several do so.

Large quantities of civil plutonium are kept today in Europe with no immediate use[56]. MOX burning capacity is already scare. New SMRs should proposes usability of MOX fuel from the start-on.

As mentioned earlier, because of the plutonium characteristics MOX fuel fabrication and logistics are more complex and consequently more expensive than for UO_2 fuel. In reactor operation MOX fuel is on the whole equivalent with UO_2 fuel. It is more constraining once discharged as regards its management in the reactor pool, handling, transport, and disposal. Associated future cost should not be neglected.

Renewed MOX fuel recycling (multi-recycling) may be problematic. MOX multi-recycling in LWRs is currently in the development phase.

[54] Potentially increasing pool saturation risk.
[55] May also have an impact on the reactor pool thermal capacity.
[56] Opportunity for generation IV reactors.

It may be decided to “save” the used MOX for deployment in future generation-4 reactors, which from a financial point of view will constitute a liability.

Alternatively used MOX can be disposed of similarly as spent uranium fuel assemblies (i.e., without reprocessing). But then ‘open cycle’ DGR precautions will be necessary, and some reprocessing benefits may be lost.

Another alternative may be to burn the separated Pu in third-party’s reactor.

The operator may choose to buy the service of burning its separated Pu in a third-party reactor. Third parties will accept using MOX instead of uranium fuel in their reactors only if the MOX fuel in its globality[57] is less costly than uranium fuel. A motivation could be future higher enriched uranium prices[58].

4.1.1.4 Uranium recycling

Reprocessed uranium (96% of the separated mass) may be reused as ERU fuel which is less complicated than MOX - but still different - in terms of fuel fabrication, logistics and spent fuel management.

RepU recycling is achieved via conversion and re-enrichment or by blending it with higher enriched fresh or used uranium, then manufacturing of enriched reprocessed uranium (ERU) fuel assemblies and burning them in LWR reactors.

Like MOX fuel assemblies, ERU fuel assemblies are structurally identical to the natural uranium fuel. They also have a comparable burnup potential and are designed with a comparable average reactivity[59].

For some operators RepU conversion (before its enrichment) or RepU blending can be obstacles today[60].

Large RepU stockpiles exist already. It can be seen as a strategic reserve for the case of disruptions in uranium supply. It would find an excellent use in generation-4 fast breeders.

Reprocessing of spent ERU fuel implies some constraints but is meanwhile routine (at La Hague).

Finally, if not used RepU becomes a low-level waste and shall be retrieved and managed by its owner.

Today, recycling appears to make again economic sense under certain conditions. In the medium term, up to 30% of the fuel requirement could be covered with thermal MOX (and ERU) fuel. The need for final repository capacity and the constraints are significantly reduced. The technologies are well known and proven and fuel assemblies with recycled Pu and U have so far shown excellent performance.

[57] Including MOX fabrication, transport, reactor handling and operations, spent fuel management and disposal.
[58] Currently enrichment capacities are a spare resource with a high “Russian component”.
[59] Slightly higher absolute enrichment is needed for equivalence to compensate for higher neutron capture in RepU.
[60] For the time being RepU conversion or RepU blending are provided only by Russia.

The introduction, or reintroduction in some European countries, of recycling requires a broad acceptance by the public as well as within the nuclear industry (as it would require a significant new build and additional reprocessing capacity).

Choosing a closed cycle policy or 'partially closed cycle' is a national strategic task.

4.2 Deep geological disposal

The long timescales over which some waste remains radioactive has led to the idea of deep disposal in underground repositories in stable geological formations. Isolation is provided by a combination of engineered and natural barriers (rock, salt, clay) and no obligation to actively maintain the facility is passed on to future generations. This is often called a 'multi-barrier' concept, with the waste packaging, the engineered repository, and the geology all providing barriers to prevent the radionuclides from reaching humans and the environment. In addition, deep groundwater is generally devoid of oxygen, minimising the possibility of chemical mobilization of waste.

The contents should be retrievable. There are reasons for keeping the waste retrievability options open – in particular, it is possible that future generations might consider the buried waste to be a valuable resource. Further it may be that mistakes occur, or the geology prediction is not accurate or underground conditions change[61] due to natural or human made catastrophes.

On the other hand, permanent closure might increase long-term security of the facility In mined repositories, which represent the main concept being pursued, retrievability can be straightforward, but any deep borehole disposal is permanent.

France's 2006 waste law says that HLW disposal must be 'reversible', which was clarified in a 2015 amendment to mean guaranteeing long-term flexibility in disposal policy, while 'retrievable' referred to short-term practicality. France, Switzerland, Canada, Japan, and the USA require retrievability. That policy is followed also in most other countries, though this presupposes that in the long-term, the repository would be sealed to satisfy safety requirements.

Deep geological disposal is the preferred option for nuclear waste management in most countries and there is much information available on different disposal concepts.

The only purpose-built deep geological repository that is currently licensed for disposal of nuclear material is the Waste Isolation Pilot Plant (WIPP) in the USA, but it does not have a licence for disposal of used fuel or HLW. Plans for disposal of spent fuel are particularly well advanced in Finland, as well as Sweden, France, and the USA, though in the USA there have been political delays. In Canada, Switzerland and the UK, deep disposal has been selected and the site selection processes have commenced.

4.2.1 Mined repositories

The most widely proposed deep geological disposal concept is for a mined repository comprising tunnels or caverns into which packaged waste would be placed. In some cases (e.g., wet rock) the waste containers are then surrounded by a material such as cement or clay (usually bentonite) to provide another barrier (called buffer and/or backfill). The choice of waste container materials and design, as well as the buffer/backfill material varies depending on the type of waste to be contained and the nature of the host rock-type available.

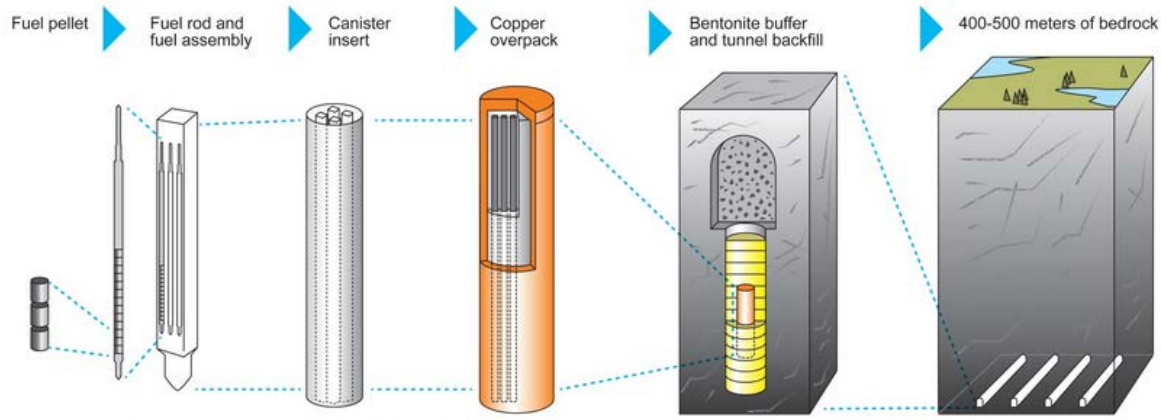
[61] In Germany billons are spent currently for the remedial measures of the Morsleben repository where the stability of the salt domes has deteriorated to a state in which collapse could occur.

Excavation of a deep underground repository using standard mining or civil engineering technology is limited to accessible locations (e.g., under land or nearshore), to rock units that are reasonably stable and without major groundwater flow, and to depths of between 250m and 1000m. The contents of the repository would be retrievable in the short term, and if desired, longer-term.

One of the major problems associated with mined repositories relates to the transport of wastes.

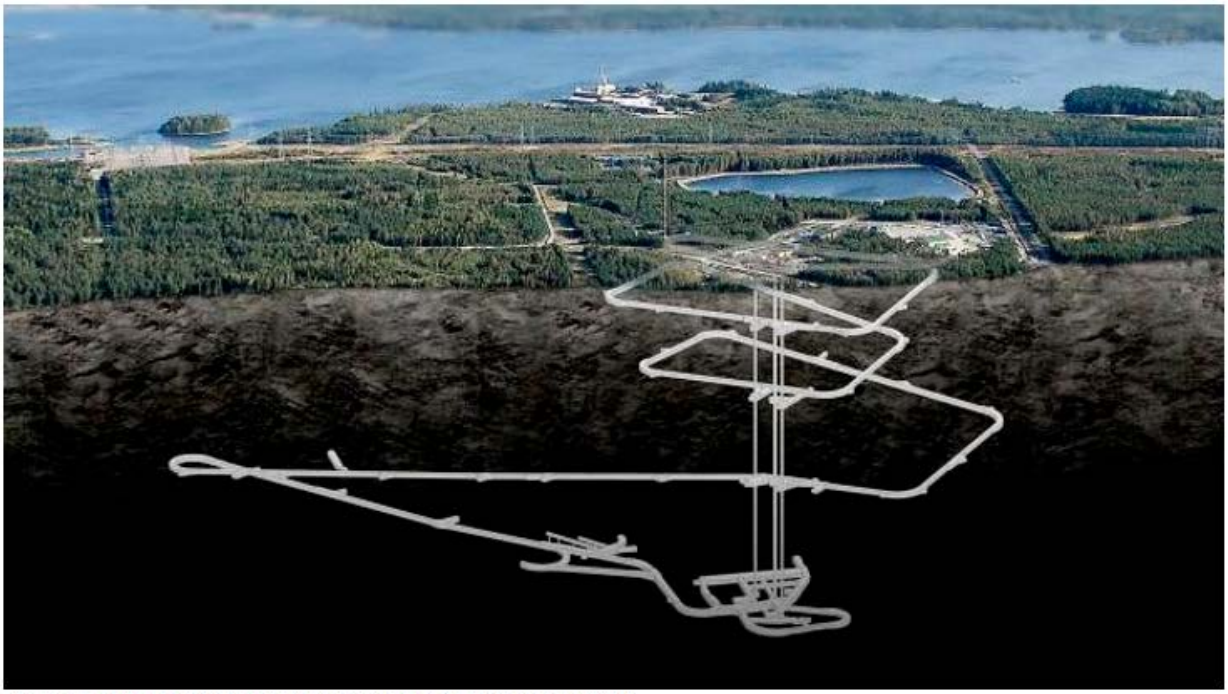
The Swedish proposed KBS-3 disposal concept uses a copper container with a steel insert to contain the spent fuel. After placement in the repository of about 500 metres deep in the bedrock, the container would be surrounded by a bentonite clay buffer to provide a very high level of containment of the radioactivity in the spent fuel over a very long time period. In June 2009, the Swedish Nuclear Fuel and Waste Management Company (SKB) announced its decision to locate the repository at Östhammar (Forsmark). The KBS-3 system consists of a central facility for interim storage and encapsulation of used nuclear fuel, a transport system for the transportation of canisters with encapsulated used nuclear fuel and a final repository facility.

Finland's repository programme is also based on the KBS-3 concept. Spent nuclear fuel packed in copper[62] canisters will be embedded in the Olkiluoto bedrock at a depth of around 400 metres. The country's nuclear waste management company, Posiva Oy, expects the repository to begin disposal operations in 2025. Its construction was licensed in November 2015.



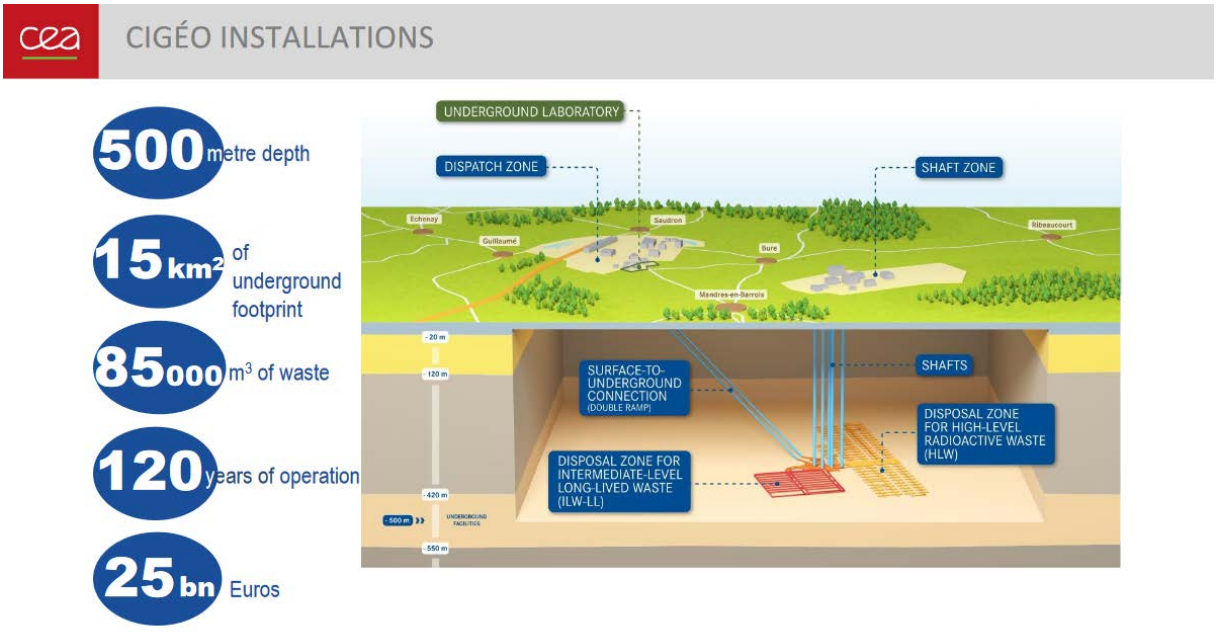
Multi-barrier disposal concept (Image: Posiva)

[62] The deposits of native (pure) copper in the world have proven that the copper used in the final disposal container can remain unchanged inside the bedrock for extremely long periods, if the geochemical conditions are appropriate (low levels of groundwater flow).



A rendering of the underground used fuel repository at Olkiluoto (Image: Posiva)

In France the current fuel cycle strategy is the mono-recycling strategy and is embedded in a long-term objective of complete closure of the fuel cycle, which is meant to be achieved by multi-recycling of spent fuels in fast-breeder reactors (FBRs). Most of France's HLW is currently vitrified and conditioned in stainless steel containers and placed in intermediate (shaft-type) storage at Orano's La Hague plant (waste derived from the reprocessing of spent fuel). Given its half-life, the law stipulates the transfer of these containers to the reversible geological disposal at ANDRA's Industrial Centre for Geological Disposal (Cigéo). Built on the boundary between the Meuse and Haute-Marne departments, Cigéo is scheduled to open in 2035. Waste will be stored in drifts hollowed out 500 metres below ground, in a stable geological environment, embedded in impermeable claystone.



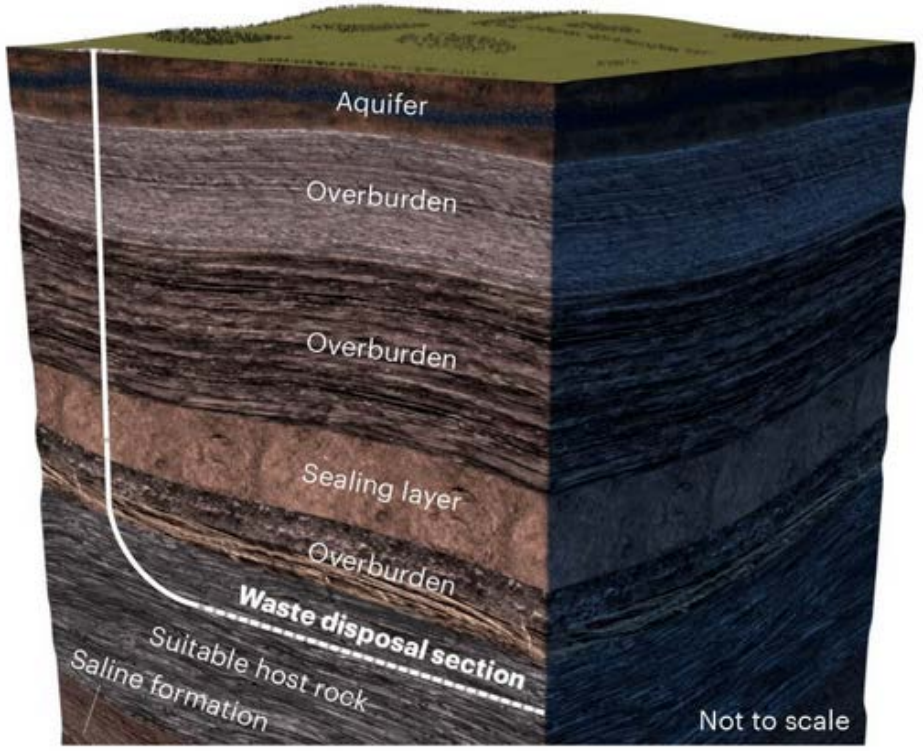
The volume of conditioned HLW from reprocessing being reduced as compared to spent fuel, the future DGR footprint will be consequently also smaller.

4.2.2 Deep borehole disposal

Deep borehole disposal (DBD) has been considered as an option for geological isolation for many years, including original evaluations by the US National Academy of Sciences in 1957 and more recent conceptual evaluations. Advances in deep drilling technology over the past 20–30 years have led to the reconsideration of deep borehole disposal.

In contrast to mined repositories, the contents would not be retrievable.

The concept consists of drilling a borehole into basement rock to a depth of up to about 5000 metres, emplacing waste canisters containing used nuclear fuel or vitrified radioactive waste from reprocessing in the lower 2000 metres of the borehole and sealing the upper 3000 metres of the borehole with materials such as bentonite, asphalt or concrete. The disposal zone of a single borehole could thus contain 400 steel canisters each 5 metres long and one-third to half a metre in diameter. The waste containers would be separated from each other by a layer of bentonite or cement. It has been suggested that vitrified HLW packages are suitable for DBD without overpacks, which may be debatable[63].



Deep borehole disposal concept – natural barriers (source: Deep Isolation)

[63] It would be prudent to allow for an overpack of a certain wall thickness.

In the DBD concept, less reliance is placed on engineered barriers, because at depths of several kilometres, the host rock is assumed to provide greater isolation and containment than for shallower geological repositories[64].

Boreholes can be readily drilled offshore as well as onshore in both crystalline and sedimentary host rocks. This capability significantly expands the range of locations that can be considered for the disposal of radioactive waste.

Compared with deep geological disposal in a mined underground repository, placement in deep boreholes may be more expensive for large volumes of waste. A large total volume, large package sizes or the need for retrievability potentially render the waste unsuitable for deep borehole disposal.

DBD could reduce or eliminate the transport issue through its potential for dispersed disposal. The footprint of an individual borehole is tiny and even for a multi-borehole array it is quite small. Boreholes need be only a few tens of metres apart. Consequently, a DBD programme could involve many small sites with only one or a few boreholes each, even extending to individual nuclear power plants disposing of their own wastes on or near site, provided nearby suitable geology. Further it is assumed that DBD may allow relatively early disposal of heat-generating wastes shortening interim storage times.

The main perceived disadvantage of DBD is the near irretrievability of the wastes. In countries where retrievability of the wastes beyond the point of closure of the repository (or borehole) is a legal or regulatory requirement DBD is not an option.

Regarding non-proliferation safeguards and the security of fissile materials, DBD could be advantageous, as recovery of packages would not be possible.

For nations with a small nuclear power programme or with no nuclear power programme but with research reactors and other nuclear R&D waste, deep borehole disposal seems promising and is likely to be more cost effective than a mined repository.

Eventually the same societal issues may emerge for siting boreholes as for siting a geological repository. The greater depth of burial, safety and availability of technically suitable sites could facilitate public and political acceptance.

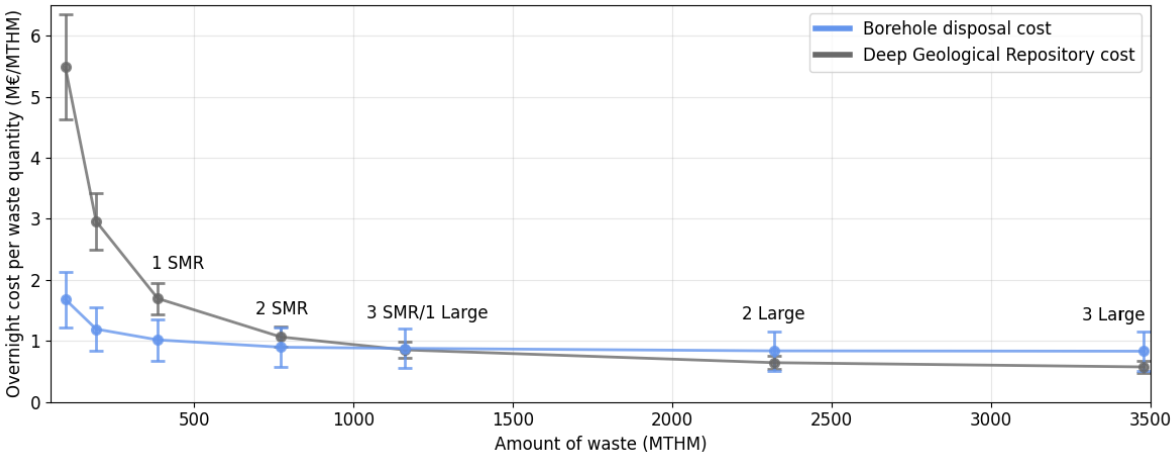
The borehole disposal compares to a conventional geological repository seems simpler to design, construct and operate and easier to demonstrate its short-term and long-term safety and its modularity. Nevertheless, future field-based demonstration projects must address key elements of borehole disposal, including deep drilling of wide-diameter holes, waste emplacement testing and seal emplacement and performance monitoring.

In a current, not finalised study the National Institute of Chemical Physics and Biophysics is analysing the potential spent nuclear fuel disposal options for Estonia and associated cost forecast. The expected overnight cost (2022) for disposal of spent nuclear fuel from the BWRX-300 small modular reactor are estimated for the beep borehole and mined repository concepts.

Preliminary comparative cost estimation let expect that for one 300MW SMR borehole disposal is noticeably cheaper than the mined deep geological repository option (400M€ vs.700M€ overnight).

[64]Nevertheless, a minimum level of engineered barriers is considered for both horizontal and vertical boreholes (corrosion-resistant alloys, cementitious materials for sealing and support matrices).

Larger amounts of waste reduce cost differences between the solutions and then make the deep geological repository cheaper (for 3 or more such SMRs).



4.2.3 Multinational repositories

Not all countries are adequately equipped to store or dispose of their own radioactive waste. Some countries are limited in area, or have unfavourable geology, and therefore siting a repository and demonstrating its safety could be challenging. Some smaller countries may not have the resources to take the proper measures on their own to ensure adequate safety and security, or they may not have enough radioactive waste to make construction and operation of their own repositories economically feasible.

It has been suggested that there could be multinational or regional repositories located in a willing host country that would accept waste from several countries. They could include, for example, use by others of a national repository operating within a host country, or a fully international facility owned by a private company operated by a consortium of nations or even an international organization. However, for the time being, many countries would not accept nuclear waste from other countries under their national laws.

The legal implications of involvement in a multinational repository cover a wide range of topics. The most prominent of these are concerned with the transfer of ownership of the waste to the host country, the nuclear liabilities regime involved, waste transportation, commitment to the completion of the project and possible damage in the event of non-performance.

The 2011 EC Waste Directive allows that two or more member states can agree to use a final repository in one of them. The export of radioactive waste for disposal in countries outside the EU is allowed only under strict conditions. Safety standards drawn up by the International Atomic Energy Agency are legally binding[65].

A multinational working group[66], whose members are nominated by the appropriate Government level organisations, is established to study the feasibility of setting up a Development Organisation (ERDO) that would implement one or more shared geological repositories in Europe.

[65] This includes an independent authority which grants licences for building repositories and checks the safety analysis for each individual repository.
[66] Estonia was a member of ERDO.

However, despite intensive international efforts, no "voluntary" EU state has yet come forward. (Earlier the establishment of international repositories in Australia and Canada were considered and abandoned.)

Co-operation among Latvia, Lithuania and Estonia is traditionally close and pragmatic. When dealing with energy security and security of supply in the framework of the Baltic Assembly and the Baltic Council of Ministers[67], a shared regional repository located in one Baltic country could be investigated. Estonia is holding the presidency in the Baltic Assembly and the Baltic Council of Ministers in 2023.

From a multinational perspective, when formulating a national policy and strategy for radioactive waste and spent fuel management it will be important to address export/import of radioactive waste, spent fuel management, radioactive waste management, and public information and participation.

However, there are almost no challenges faced by multinational disposal initiatives that are not also faced by purely national disposal programmes in democratic countries.

4.2.3.1 Fuel leasing

Fuel leasing is an alternative concept whereby the utility leases its fabricated fuel from a supplier, probably in another country, and after it has been used that supplier takes it back. This concept is not yet in use except for some very limited applications, mainly for Russian-built nuclear power plants in NPT non-weapons states (e.g., Bangladesh, Turkey, Iran). The supplier would then add the leased used fuel to its own larger stocks to be stored for later disposal or reprocessing and recycling, in which case the valuable components would belong to the fuel supplier/leaser.

[67] The Baltic Councill of Ministers, an institution for governmental co-operation between Latvia, Lithuania and Estonia charged with ensuring the continuity of co-operation at the executive level of the states.

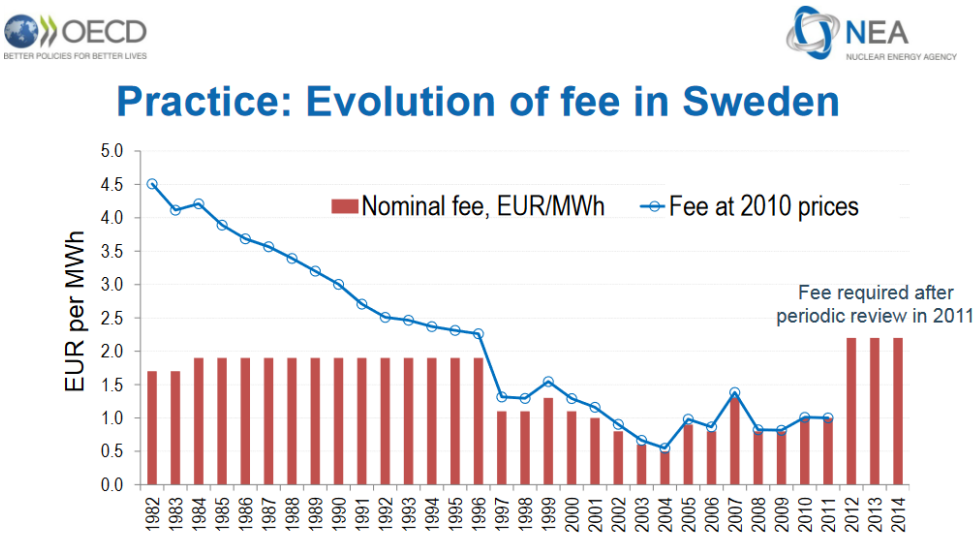
5 Options for financing spent fuel and HLW management Disposal of spent fuel and financial aspects relating to the safety of radioactive waste management

A fundamental prerequisite for implementing a sustainable waste disposal programme is the provision of financial resources to cover the costs of the programme.

Financial liabilities are the future costs resulting from past or current activities and need to be identified and correctly assessed so that provisions are made to cover the expenditures to discharge those liabilities.

Reliable methods for estimating the cost of the disposal programme are therefore needed to ensure that adequate funding is available and will be maintained until the disposal programme's completion, including post- closure monitoring and institutional control. Cost estimates will also assist decision makers and engineers in the development of the disposal concept, as they can quantify the cost impact of certain design choices. It is therefore necessary that these cost estimation methods are implemented early in the disposal programme. To guarantee adequate financing, periodic assessments of the radioactive waste management costs are essential.

Another aspect of a disposal programme is the potentially long-time span between when funding for the project is acquired and when the actual expenses arise. The most common principle is that the waste producer is responsible for all the costs of the waste management, including disposal. The cost estimates therefore must take account of the disposal project long before its implementation to assess the required funding including provisions for risks and uncertainties. It is therefore important to establish appropriate financial arrangement and careful management of the accrual of the collected funds. The deployment of deep geological repositories for radioactive waste will reduce uncertainties as shown below for the case of Sweden.



Initial values covered substantial levels of uncertainties, which could be gradually reduced as more accurate knowledge of costs had been gained through further advancements of the programme.

Source: OECD/NEA, *Economics of the Back-end of the Nuclear Fuel Cycle*, 2013.

For nuclear power generation, this cost is generally included in the production cost and should be covered by power production revenues and the collection of funds needs to be undertaken in advance of plant closure.

5.1 Financing spent fuel and HLW management

The most common funding sources are the waste producers and the State. Selecting the most suitable funding mechanism depends on many factors, such as the nuclear strategy and policy of the State, the disposal option, the legislative background and the institutional framework.

The funding mechanism for states with nuclear power plants should consider the relevant types or categories of waste, the associated time scales and weather to operate a single fund for decommissioning and waste management or separate funds[68]. The case of borehole disposal maybe particular, more straight forward in terms of funding as regards cost and time scale.

5.1.1 The waste producer

Following the 'polluter pays' principle the waste producer will provide the necessary financial resources.

For nuclear power plant operators, a waste management levy based on the amount of electricity generated is one approach. Another way is to charge the waste producers a 'once and for all' charge when the waste is transferred to the waste management operator (WMO), by applying a waste tariff system for different waste categories. A combination of both systems can also be applied for electricity and institutional waste producers. Applying a tariff on the waste volume will incite waste producers to reduce their waste volume. The tariff design needs other factors, such as activity or toxicity. Another possible waste tariff system is based on selling space in the disposal facility. Waste producers can also be charged directly for any disposal services rendered. This can be applied for disposal facilities in operation and is therefore more commonly used for the disposal of LLW and ILW rather than for HLW or spent fuel, for which disposal often lies in the future.

The waste producers' contributions can be collected in a fund (internal or external[69]), or they can be set aside as provisions on the balance sheets of the waste producers.

Some examples: The most common mechanism adopted for the accrual of funds are levies on nuclear electricity. E.g., 0.1 ¢/kWh in the US or 0.14 ¢/kWh in France or 0.2 ¢/kWh in Romania. Or in other cases, waste producers pay lump sums (e.g., in Korea) or proportionally to the volumes of waste produced (e.g., in Belgium).

5.1.2 The state

In some cases, the state is the only available source of funding.

This can happen at the early stage of the waste disposal programme when the WMO is being set up and when waste management systems and infrastructures are being developed or resources for waste management might not be available or when the waste owner is unknown or no longer exists.

Certain circumstances may also justify such an approach on the basis that all citizens have an interest in the safe management and disposal of the waste.

[68] E.g., nuclear utilities in Switzerland make separate payments into two separate funds: decommissioning and waste management.

[69] External fund: payments made into a fund that is held outside the organization, often within governmental bodies or administered by a group of independent trustees. A 3rd party administering funds promotes transparency, insolvency protection and enhances confidence.

The State can support the startup phase of the WMO when another funding mechanism has not been developed. The State can also financially support the disposal programme by providing guarantees and enable the WMO to get financial resources through bank loans or other financial instruments.

Some examples: Lithuania (insufficient funds), Germany (Federal Government since 2017 in return of a 24B€ payment by the nuclear operators), Hungary (waste producers and state paying interests keeping up the fund value).

5.1.3 Contribution plan

A fund contribution plan needs to be defined, the aim of which is to ensure that the fund will be sufficient to cover the disposal costs. The two components to the contribution plan for the fund are its target value (i.e., specifying how much funding will be needed) and the contribution schedule of the fund.

The estimated overnight cost in determining the target value of the fund must be escalated over the time[70] value of money.

One of the most significant uncertainties concerns the schedule or planning for the disposal programme because of the impact the time schedule has on the discounted cost. The uncertainties around the discounted costs should be taken into account in the management of uncertainties and risks.

Due to initial assumptions and uncertainties, it will be necessary over the time to revise the contribution plan periodically.

Not to discount future costs could be a more conservative approach, as it leads to a faster accumulation of provisions. In such case the amount needed to cover a liability is gathered as soon as the liability appears. However, it renounces the opportunities which could be exploited with an efficient investment strategy.

Which method is chosen (i.e., whether to discount the cost or not) often depends on the prevailing legislation and accounting practices, e.g., If provisions are tax deductible or interest generated by the provisions are subject to taxation.

Some examples: In Finland the funding system is based on undiscounted costs. Money deposited into the fund gains interest every year, which reduces the annual amount of new funding required. In Belgium the fees are increased above inflation each year by a constant interest rate fixed at 1% whereas in Canada the funding requirement is based on the discounted future cost of the project.

5.1.4 Contribution schedule

After the target fund value is determined, the schedule by which the contributions will meet the target needs to be specified. This requires knowledge of the total waste inventory that will be disposed of the duration of the waste producing activities or the amount of (future) nuclear electricity generation. This information might not be available and might need to be assumed.

In cases where annual contributions will be paid into the fund based on an assumed operating lifetime of the facility generating radioactive waste, the most straightforward contribution schedule

[70] Cost discounting: money available today is worth more than in the future due to its capacity to earn money through interest and investment.

is a constant one. This means that the contribution will be the same for every year of the facility lifetime.

Facility lifetime might be shorter or future waste volumes higher than anticipated and potentially resulting in insufficient funding. The risk of insufficient funding can be reduced by a 'front end loaded' contribution schedule. The fund could be collected over a shorter period or by a prepayment. A sensible approach could be to follow a front end loaded schedule to fund the fixed costs of the disposal programme while applying a constant one for the costs that vary with the amount of waste.

5.1.5 Fund management

The waste producers' contributions can be collected in a fund or the anticipated disposal costs can be included on the company's balance sheet as a liability.

A fund is commonly set up for funding costs that lie far in the future, such as the disposal of HLW or the post-closure phase of the disposal project. The time span between the acquisition of the funding and the time when the actual expenses arise offers the possibility of taking advantage of the time value of money in the fund. However, it also entails the risk of mismanagement or even misuse of the fund.

Different models of fund ownership exist, and a distinction is made between internal funds (managed by the waste producers) and external funds (managed by an organization separate and independent from the waste producer, like WMO or government).

External fund offers a transparent model that facilitates verifying whether the financial resources are available, adequate, and used for the intended purpose. It also offers protection against a loss of funding in the case of operator bankruptcy.

In the case of an internal fund, it can be a segregated or ring-fenced fund, which means that it can only be used for the purpose for which it is set aside.

A distinction can be made between a centralized fund for the entire industry or for decentralized funds, of which are as many as there are operators.

Some examples: In Canada owners of spent fuel have been required to establish segregated funds for the long-term waste management of the spent fuel since 2002. In France, too the waste management fund is an internal segregated fund and reviewed by the government. In Switzerland for the costs arising after the closure of nuclear power plants, the owners set aside provisions in an external fund managed by the commission STENFO (Stilllegungs- und Entsorgungsfonds), which is under the supervision of federal authorities. Costs arising before closure are paid directly by the owners and corresponding internal provisions are made.

5.1.6 Investment strategy

The growth of the fund depends on the investment strategy.

The fund resources may be invested in industrial or public activities, including company shares and corporate or governmental bond. Sufficiently conservative strategies are recommended. There is the risk that the return on the investment might be lower than anticipated when discounting the disposal programme's cost. It will be necessary to assume a return on investment that is realistic for low-risk investments.

Another risk that needs to be considered when investing the fund resources is the liquidity risk at the (uncertain) time when the resources need to be available.

Some examples: In Finland a maximum of 75% of the Nuclear Waste Management Fund can be lent back to the operators, who in return need to pay a government fixed amount of interest. In the Czech Republic the funds at the Nuclear Account may be invested in financial markets in liquid government bonds, the Czech National Bank and securities of emitters selected by the Ministry of Finance.

5.1.7 Capital growth

An important financial aspect of a disposal programme is the potentially long-time span between when funding for the project is acquired and when the actual expenses will arise. Such long timespan can be an excellent opportunity to allow sufficient capital growth of the fund, to cover all expenses until the disposal programme's completion. Especially for countries with a small nuclear fleet long-time spans could avoid the state's (taxpayer's) backing.

Some examples: In The Netherlands COVRA is responsible for the capital growth of the fund (over 100 years). The money in the fund is put in safe investments (e.g., government bonds) which must be approved of by the minister of economic affairs.

5.2 Weighing 'open cycle' vs. 'closed cycle' impact

The national policy regarding the management of spent fuel may define spent fuel as waste to be disposed of directly or as a future resource with the spent fuel being (re)utilized through reprocessing.

Modelling the two methods over long periods is unconclusive from a financial point of view. Overnight cost simulations hold the balance or incline to one option or the other depending on the respective study presumptions. Long-term investigation results are mainly driven by risk assumptions.

Implementation of an 'open cycle' or 'closed cycle' strategy will have an impact on the cost timing and spending breakdown of a final disposal infrastructure. Under some circumstances it may also affect the interim storage necessities.

5.2.1.1 Open cycle

The fixed cost for provisioning of a mined deep geological repository, i.e., all infrastructures except for the number tunnels or caverns into which packaged fuel assemblies will be placed, depends in a lower degree on the quantities of spent fuel to be disposed of. One major cost factor is the availability at the disposal site at the time of disposal of adapted equipment for handling (unpacking and conditioning/repacking) of spent fuel assemblies[71]. Such large size "hot cells" are an important cost factor and will be required regardless of the spent fuel quantity to be conditioned for DGR disposal. The "lion's share" of expenses will occur late in time.

5.2.1.2 Closed cycle

A "reprocessing only" strategy would eventually result in having to manage only standardised conditioned HLW[72] and reducing the volume by a factor of 5.

Earlier, non-negligible, expenses for used fuel reprocessing will be required during the operational lifetime of the nuclear power plant. Depending on the appointing of responsibilities for spent fuel

[71] Important cost factors are above- and under-ground facilities, for operating the final storage site and, for repository closure. Minor costs accrue for planning, preparation work, underground maintenance, decommissioning.

[72] And LL-ILW.

management, such cost will be absorbed either by the NPP operational budgets or the final disposal funds. The strongest arguments against reprocessing today are economic.

Other than for sustainability reasons, the reprocessing strategy may be driven by the opportunity of reducing financial risks regarding future HLW management cost.

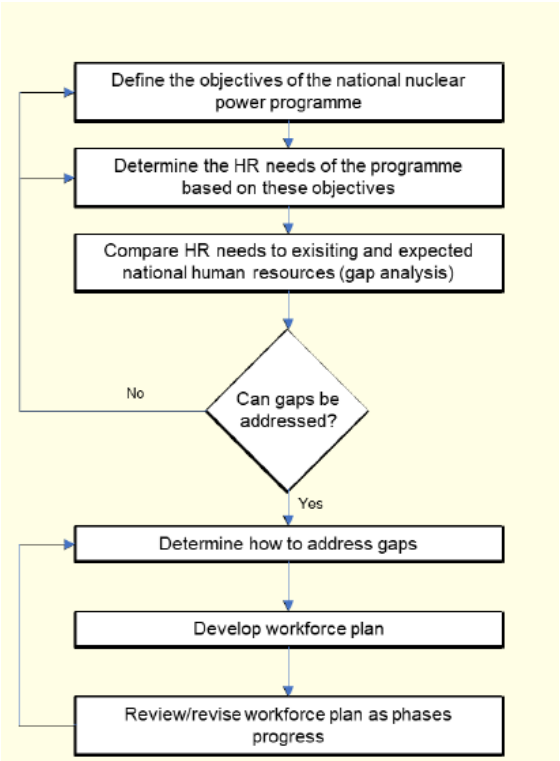
Earlier expenses in return for reduced risks may lead to less total expenses and funding necessities.

6 The human resource and other infrastructure development needs associated with radioactive waste management for a nuclear power programme

Estonia's pre-existing legal and regulatory framework and the current organisational set-up are addressed in paragraph 2.

Following the development of the legal framework for nuclear activities and to ensure its implementation existing entities may be redefined and enlarged (e.g., Environmental Board, A.L.A.R.A.) or/and new organisations will be established.

Regarding the workforce planning IAEA proposes the following process:



Workforce Planning Process by IAEA

6.1 The independent regulatory body

The future new nuclear law shall establish the independent regulatory body having the authority to determine all matters relating to the control and supervision of the nuclear sector in relation to safety, security, and safeguards.

The regulatory body's manpower dedicated to long term waste management issues could be build up gradually when developing and implementing the national strategy for the back-end fuel cycle and radioactive waste management.

A qualified and capable workforce will be necessary for staffing of the future regulatory body and waste agency. For this the UAE "two-track" human resources strategy may serve as a model:

- Recruitment of staff with nuclear experience and skills from outside the country to direct and support the “quick launch” of the programme,
- Development of nationals to take increasingly responsible positions in the regulatory body, while retaining a cadre of international experts.

Competence areas, high-level sourcing strategies and the number of FTEs for phase 1 must be identified. Existing competence in Estonia should be utilized to a high degree. Recruitment (repatriation) of Estonian nationals specialised abroad in nuclear related areas should also be considered.

The need for highly specialized nuclear knowledge is limited. The required areas of competence, if strictly scientific and technical matters are outsourced, are in licensing & regulatory & legal affairs, public & communication, safeguards, funding and financing, document control, import/export control, strategic safety & security, radiation protection and strategic waste issues, project (oversight) & quality management.

In areas where the competencies of regulatory body and/or waste agency are not sufficient dedicated resources shall be provided by including research facilities, universities, laboratories and third parties, such as technical support organisations (TSO) and specialist consultants.

The recruited staff should be “nuclear aware” with a minority of ~10% of experienced nuclear specialists. All new recruits shall undergo nuclear specific training for their area of competence with a focus on nuclear quality culture.

The strategy should be strengthened through a programme for education, ongoing training and development opportunities in international cooperation.

A scholarship programme for selected high school students that should be granted sponsorships to study abroad for degrees in nuclear related disciplines is recommendable.

6.2 The waste management agency

The nuclear law will also establish the legal framework[73] under which the waste management will operate and establish and define the responsibilities of a national waste agency.

A.L.A.R.A., the currently existing waste agency could be germ of the new national waste agency with enlarged responsibilities for the waste generated by the new build SMR programme.

In this case too, qualified and capable workforce will be necessary and could be enlarged gradually when implementing the national strategy for the back-end fuel cycle and radioactive waste management.

6.3 The D&D and waste management fund

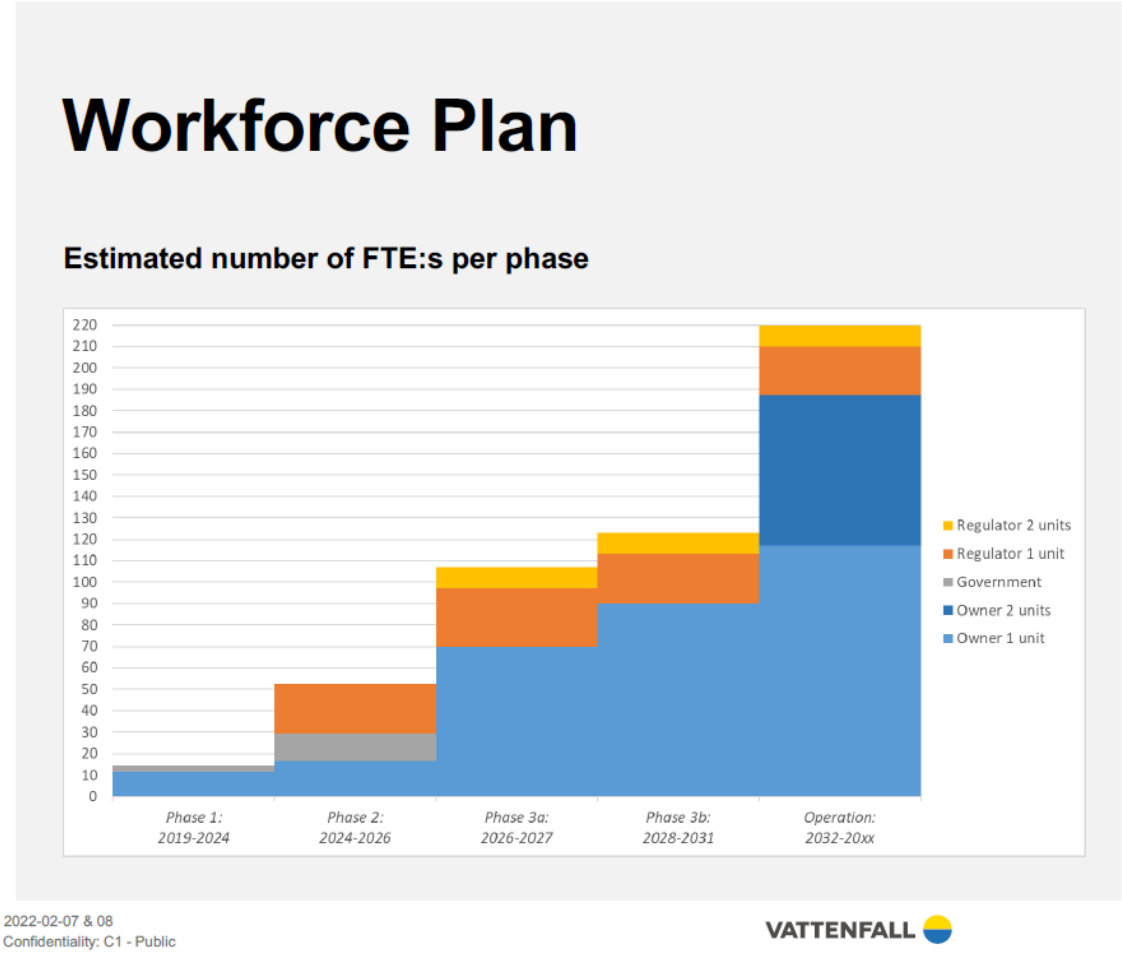
Further, the new nuclear law will ensure the financing mechanism of the regulatory body, the waste agency and decommissioning and disposal funds.

The regulatory body and waste agency should be financed by the state, or the state should have/implement control over financing if monies are collected from nuclear operators. A relatively important percentage of the budget shall be allocated to outsourcing.

Appropriate institutions and funding & financially proficient staff shall be appointed concerning the waste management fund, depending on the chosen model of fund ownership (see paragraph 5.1.5.

[73] Derived from the national policy which defines the principles for the management of radioactive waste.

Under certain circumstances this task could be provided by the waste management agency. D&D funding could be part of the waste management fund or set up separately.



Source: [23] Eva Simonsen, HR Development Strategy for Estonia

6.4 Development in cooperation with other countries or international organizations

6.4.1 Support in regulatory activities

Estonia may benefit from strong relations with the regulatory authorities in the vendor country of origin and the country of the FOAK (first-of-a-kind) SMR. Cooperation agreements could be envisaged covering:

- Exchange of information on the regulatory framework, licensing process and safety assessments of the reference plant,
- Workshops & technical meetings,
- Access to experts,
- Support for vendor inspections,
- Support for training and staff secondments.

To augment its internal resources for reviewing licence applications external Technical Support Organizations (TSOs) located in the USA, Canada and Europe could be brought in.

The geographical and cultural proximity suggests cooperation with STUK, the Finnish Radiation and Nuclear Safety Authority[74].

6.4.2 **Dual-track approach** (see also paragraph 4.2.3 “Multinational repositories”)

Every country using nuclear technologies generates wastes that require geological disposal. A deep geological repository (DGR) is the most complex waste management facility to implement. For almost all countries, target dates for operation of a national DGR lie decades into the future. This allows ample time for evaluation in parallel of the shared DGRs, that are one focus of the ERDO Association.

ERDO members are developing both options until the optimum solution for each country becomes apparent – this is the Dual-Track approach. It allows countries to keep options open whilst fulfilling their obligations to the IAEA and the EC.

Co-operation among Latvia, Lithuania and Estonia[75] is traditionally close and pragmatic. A shared regional repository located in one Baltic country could be investigated.

[74] There are calls to develop STUK’s regulatory knowhow specifically suitable for SMRs as an export opportunity for Finland.

[75] Estonia is holding the presidency in the Baltic Assembly and the Baltic Council of Ministers in 2023.

7 Recommendations for the development of the national radioactive waste management policy

7.1 National waste management policy and strategy framework

The EU's Radioactive Waste and Spent Fuel Management Directive 2011/70/Euratom requires that all EU countries have a national policy for spent fuel and radioactive waste management and that they draw up and implement national programmes for the management of these materials. The programmes should cover all types of spent fuel and radioactive waste under EU countries' jurisdiction and all stages of spent fuel and radioactive waste management from generation to disposal.

It will be necessary to establish the national policy and a technical strategy, or strategies, for the management of their radioactive waste. The two planning concepts are linked; the policy establishes the principles for radioactive waste management and the strategy contains the approaches for the implementation of the policy. For this reason, they should be closely coordinated: policy is mainly the responsibility of the national government and may become codified in the national legislative system while strategy is usually established by the relevant waste owner or nuclear facility operator, or by the government.

The national policy defines the goals and principles of waste management, and the strategy describes the approach for implementing it. It should include a set of goals or requirements to ensure the safe and efficient management of spent fuel and radioactive waste in the country.

7.2 The national policy

Estonia has already national policy and strategy document for the present rad-waste management needs. For a new nuclear program, it shall be replaced or updated.

The new national policy should identify:

- The government organization(s) responsible for establishing the legislative and regulatory framework,
- The relevant regulatory body (to enforce the implementation of the regulations on spent fuel and radioactive waste management),
- The organization(s) responsible for ensuring that radioactive waste is safely managed (normally the licencee),
- The organization(s) devoted to coordinating or overseeing radioactive waste management, responsible for the long-term management of spent fuel and radioactive waste, and for radioactive waste for which no other organization has responsibility,
- Other governmental bodies may have roles in the process, for example, government organizations concerned with environmental protection and the transport of radioactive material as well as local governmental organizations.

Several possibilities may be considered for the policy regarding the management of spent fuel:

- The national policy may consider spent fuel as a resource with the spent fuel being utilized through reprocessing (nationally or internationally) and the resulting HLW being disposed of afterwards,
- The national policy may regard the spent fuel as a waste to be disposed of directly, or

- The national policy may require that spent fuel is returned to the supplier (e.g., in fuel leasing). This last option is directly affected by the import/export policy of both the user and supplier countries,
- The national policy may allow export or import of spent fuel and/or nuclear waste including HLW in the framework of a multi-national disposal programme.

It is important to develop the waste management policy and strategy in consultation with all stakeholders in waste management, which comprise governmental bodies and regulators, the waste producers and organizations responsible for managing the waste (e.g., technical support organizations), local communities or the general public, non-governmental organizations, research institutions and advisory and consultative bodies.

7.3 The national strategy

Once a policy has been established, the approach to implement it is described in the waste management strategy. The understanding of available waste management options is necessary, among other.

The waste management options concern the technical options to collect, characterize and segregate the waste, treat, and condition it, and then store it. For HLW or spent fuel storage is an interim solution pending final disposal.

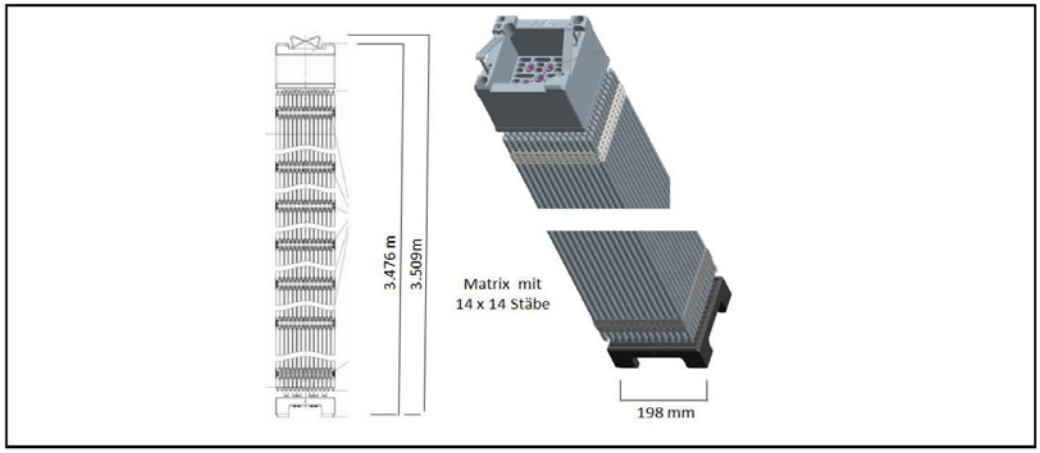
Knowledge of waste management strategies in other countries can be useful, as these can give guidance and serve as an example.

Prior to developing a waste management strategy, the legal framework under which the waste management will operate needs to be established, together with the regulatory framework defining how the waste management activities shall be regulated. The government is responsible for establishing this framework and needs to designate an independent regulatory body to enforce the waste management regulations.

Appendix 1: Swiss KK Beznau (KKB) detailed, representative PWR 14x14 UO_2 spent fuel characteristics and isotopes

MIRAM 14 - Basisszenarium MIRAM 14	Referenzjahr: 2075
BE-B-UO2-U-HAA - abgebrannte UO2-Brennelemente KKB unverpackt	

SCHEMATISCHE DARSTELLUNG DES REPRÄSENTATIVEN GEBINDES



AUFBAU DER ABFALLSORTE AUS ABFALLGEBINDETYPEN (AGT)

Anzahl			*pro Gebinde*	
Gebinde	*AGT Code*	*Bezeichnung*	*m^3*	*kg*
128	**J-B-950001**	@ BE KKB (UO2): 3.36% 35.9 GWd/t	1.38E-01	4.87E+02
369	**J-B-950002**	@ BE KKB (UO2): 3.71% 43.5 GWd/t	1.38E-01	4.87E+02
194	**J-B-950003**	@ BE KKB (UO2): 4.54% 52.4 GWd/t	1.38E-01	4.87E+02
748	**J-B-950004**	@ BE KKB (UO2): 4.50% 55.0 GWd/t	1.38E-01	4.87E+02
242	**J-B-950005**	@ BE KKB (UO2): 4.50% 38.3 GWd/t	1.38E-01	4.87E+02
1681	**BE-B-UO2-U-HAA**	abgebrannte UO2-Brennelemente KKB unverpackt	**1.38E-01**	**4.87E+02**

MIRAM 14 - Basisszenarium MIRAM 14	Referenzjahr: 2075
BE-B-UO2-U-HAA - abgebrannte UO2-Brennelemente KKB unverpackt	

MATERIALKENNDATEN FÜR DAS REPRÄSENTATIVE GEBINDE

Materialinventar und Herkunft

Material	*Masse [kg]*		*Herkunft und Masse [kg]*				
		*	*C*	*D*	*E*	*F*	*G*
			Abfallprodukt	*Behälter*	*Füllmaterial*	*Einbauten*	*Leerraum*
Inconel 718	**7.36E-01**	M				7.36E-01	
Luft	**9.52E-02**	A					9.52E-02
Plutonium(IV)oxid PuO2	**3.86E+00**	A	3.86E+00				
Spalt/Akt.produktoxide BS	**2.06E+01**	A	2.06E+01				
Stahl 1.4541	**1.22E+01**	M				1.22E+01	
Stahl 1.4571	**1.97E+00**	M				1.97E+00	
Uran(IV)oxid UO2	**3.44E+02**	A	3.44E+02				
Zircaloy-4	**1.03E+02**	M		8.74E+01		1.60E+01	
Total	**4.87E+02**		**3.69E+02**	**8.74E+01**	**0.00E+00**	**3.10E+01**	**9.52E-02**
* total anorganisch/nichtmetallisch	**3.69E+02**	A	3.69E+02				9.52E-02
metallisch	**1.18E+02**	M		8.74E+01		3.10E+01	

Oberflächen/Massen-Verhältnisse von Metallen

Material	*m^2*	*kg*	*m^2/kg*
Inconel 718	1.77E-01	7.36E-01	2.41E-01
Stahl 1.4541	3.12E-01	1.22E+01	2.55E-02
Stahl 1.4571	5.03E-01	1.97E+00	2.55E-01
Zircaloy-4	4.25E+01	1.03E+02	4.11E-01

MIRAM 14 - Basisszenarium MIRAM 14	Referenzjahr: 2075
BE-B-UO2-U-HAA - abgebrannte UO2-Brennelemente KKB unverpackt	

RADIOLOGISCHE KENNDATEN FÜR DAS REPRÄSENTATIVE GEBINDE

Nuklidinventar

Nuklid	*Aktivität [Bq]*		*Nuklid*	*Aktivität [Bq]*		*Nuklid*	*Aktivität [Bq]*		*Nuklid*	*Aktivität [Bq]*	
	Mittel	*Maximal*		*Mittel*	*Maximal*		*Mittel*	*Maximal*		*Mittel*	*Maximal*
H-3	5.2E+11	1.0E+12	I-129	4.7E+08	9.3E+08	Bi-208	1.4E+03	2.7E+03	U-235	2.0E+08	3.9E+08
Be-10	3.7E+06	7.4E+06	Cs-134	1.0E+08	2.0E+08	Bi-210	6.3E+04	1.3E+05	U-235M	4.1E+12	8.1E+12
C-14	2.0E+10	4.0E+10	Cs-135	8.8E+09	1.8E+10	Bi-210M	3.7E+02	7.4E+02	U-236	4.6E+09	9.1E+09
Si-32	5.0E+04	1.0E+05	Cs-137	4.7E+14	9.3E+14	Bi-211	4.7E+05	9.3E+05	U-237	2.7E+09	5.5E+09
P-32	5.0E+04	1.0E+05	Ba-133	2.2E+08	4.4E+08	Bi-212	2.9E+08	5.8E+08	U-238	3.7E+09	7.5E+09
Cl-36	3.3E+08	6.6E+08	Ba-137M	4.4E+14	8.8E+14	Bi-213	1.0E+04	2.0E+04	U-240	5.8E+03	1.2E+04
Ar-39	2.1E+09	4.1E+09	La-137	1.6E+05	3.3E+05	Bi-214	1.5E+05	2.9E+05	Np-236	4.7E+04	9.4E+04
Ar-42	2.5E+02	5.0E+02	La-138	3.1E+00	6.2E+00	Po-210	6.2E+04	1.2E+05	Np-237	6.6E+09	1.3E+10
K-40	9.2E+03	1.8E+04	Nd-144	2.9E+01	5.8E+01	Po-211	1.3E+03	2.6E+03	Np-238	5.5E+08	1.1E+09
K-42	2.5E+02	5.0E+02	Pm-145	3.4E+07	6.7E+07	Po-212	1.9E+08	3.7E+08	Np-239	5.2E+11	1.0E+12
Ca-41	4.4E+07	8.8E+07	Pm-146	5.2E+06	1.0E+07	Po-213	1.0E+04	2.0E+04	Np-240	6.4E+00	1.3E+01
Fe-55	1.2E+08	2.4E+08	Pm-147	3.1E+09	6.2E+09	Po-214	1.5E+05	2.9E+05	Np-240M	5.8E+03	1.2E+04
Co-60	2.3E+10	4.7E+10	Sm-146	4.0E+03	8.1E+03	Po-215	4.7E+05	9.3E+05	Pu-236	2.8E+04	5.5E+04
Ni-59	2.6E+10	5.2E+10	Sm-147	9.0E+04	1.8E+05	Po-216	2.9E+08	5.8E+08	Pu-238	4.1E+13	8.1E+13
Ni-63	2.4E+12	4.7E+12	Sm-151	3.6E+12	7.2E+12	Po-218	1.5E+05	2.9E+05	Pu-239	4.1E+12	8.1E+12
Se-79	1.3E+09	2.5E+09	Eu-152	5.7E+09	1.1E+10	At-217	1.0E+04	2.0E+04	Pu-240	7.6E+12	1.5E+13
Kr-81	4.4E+05	8.8E+05	Eu-154	1.3E+12	2.6E+12	At-218	2.9E+01	5.9E+01	Pu-241	1.1E+14	2.2E+14
Kr-85	4.1E+12	8.1E+12	Eu-155	1.8E+10	3.5E+10	Rn-219	4.7E+05	9.3E+05	Pu-242	3.9E+10	7.9E+10
Rb-87	3.7E+05	7.3E+05	Tb-157	3.3E+06	6.6E+06	Rn-220	2.9E+08	5.8E+08	Pu-243	7.9E+03	1.6E+04
Sr-90	3.1E+14	6.2E+14	Tb-158	4.6E+05	9.3E+05	Rn-222	1.5E+05	2.9E+05	Pu-244	5.8E+03	1.2E+04
Y-90	3.1E+14	6.2E+14	Ho-163	4.8E+05	9.5E+05	Fr-221	1.0E+04	2.0E+04	Pu-246	2.5E+00	5.0E+00
Zr-93	3.3E+10	6.6E+10	Ho-166M	1.5E+08	3.1E+08	Fr-223	6.4E+03	1.3E+04	Am-241	5.7E+13	1.1E+14
Nb-91	7.0E+02	1.4E+03	Tm-171	1.7E+03	3.3E+03	Ra-223	4.7E+05	9.3E+05	Am-242	1.1E+11	2.3E+11
Nb-92	5.6E+01	1.1E+02	Lu-176	1.2E+00	2.5E+00	Ra-224	2.9E+08	5.8E+08	Am-242M	1.1E+11	2.3E+11
Nb-93M	2.9E+10	5.8E+10	Hf-182	4.2E+02	8.4E+02	Ra-225	1.0E+04	2.0E+04	Am-243	5.2E+11	1.0E+12
Nb-94	7.8E+09	1.6E+10	Ta-182	4.2E+02	8.4E+02	Ra-226	1.5E+05	2.9E+05	Am-246	2.5E+00	5.0E+00
Mo-93	1.5E+07	2.9E+07	Re-187	1.1E+02	2.2E+02	Ra-228	1.2E+01	2.5E+01	Cm-242	9.5E+10	1.9E+11
Tc-97	1.3E+03	2.6E+03	Os-194	2.8E+02	5.5E+02	Ac-225	1.0E+04	2.0E+04	Cm-243	1.0E+11	2.0E+11
Tc-98	1.2E+05	2.4E+05	Ir-192	3.1E+04	6.1E+04	Ac-227	4.7E+05	9.3E+05	Cm-244	8.8E+12	1.8E+13
Tc-99	2.3E+11	4.6E+11	Ir-192M	3.1E+04	6.1E+04	Ac-228	1.2E+01	2.5E+01	Cm-245	8.0E+09	1.6E+10
Ru-106	4.7E+01	9.4E+01	Ir-194	2.8E+02	5.6E+02	Th-227	4.6E+05	9.2E+05	Cm-246	2.0E+09	4.0E+09
Rh-102M	6.1E+00	1.2E+01	Pt-193	1.3E+09	2.6E+09	Th-228	2.9E+08	5.8E+08	Cm-247	7.9E+03	1.6E+04
Rh-106	4.7E+01	9.4E+01	Tl-204	4.1E+05	8.2E+05	Th-229	1.0E+04	2.0E+04	Cm-248	3.0E+04	5.9E+04
Pd-107	2.0E+09	4.0E+09	Tl-206	3.7E+02	7.4E+02	Th-230	1.1E+07	2.2E+07	Cm-250	3.9E+00	7.8E+00
Ag-108	1.3E+08	2.6E+08	Tl-207	4.6E+05	9.3E+05	Th-231	2.0E+08	3.9E+08	Bk-250	1.4E+00	2.8E+00
Ag-108M	1.5E+09	3.0E+09	Tl-208	1.0E+08	2.1E+08	Th-232	1.4E+01	2.9E+01	Cf-249	3.6E+05	7.2E+05
Cd-113M	4.6E+08	9.2E+08	Tl-209	2.2E+02	4.4E+02	Th-234	3.7E+09	7.5E+09	Cf-250	1.2E+05	2.4E+05
Sn-121	8.8E+10	1.8E+11	Tl-210	3.1E+01	6.1E+01	Pa-231	6.3E+05	1.3E+06	Cf-251	2.0E+04	4.1E+04
Sn-121M	1.1E+11	2.3E+11	Pb-205	8.7E+01	1.7E+02	Pa-232	7.5E+01	1.5E+02	Cf-252	7.1E+00	1.4E+01
Sn-126	9.2E+09	1.8E+10	Pb-209	1.0E+04	2.0E+04	Pa-233	6.6E+09	1.3E+10	Cf-254	1.6E+03	3.1E+03
Sb-125	3.4E+08	6.8E+08	Pb-210	6.3E+04	1.3E+05	Pa-234	3.7E+09	7.5E+09			
Sb-126	1.3E+09	2.6E+09	Pb-211	4.7E+05	9.3E+05	U-232	2.8E+08	5.7E+08	Summe α	1.2E+14	2.4E+14
Sb-126M	9.2E+09	1.8E+10	Pb-212	2.9E+08	5.8E+08	U-233	1.9E+06	3.8E+06	Summe βγ	1.7E+15	3.3E+15
Te-125M	8.2E+07	1.6E+08	Pb-214	1.5E+05	2.9E+05	U-234	2.3E+10	4.6E+10	**Total**	**1.8E+15**	**3.5E+15**

MIRAM 14 - Basisszenarium MIRAM 14	Referenzjahr: 2075
BE-B-UO2-U-HAA - abgebrannte UO2-Brennelemente KKB unverpackt	

Spaltstoffinventar

Nuklid	*Aktivität [Bq]*		*Masse [g]*	
	Mittel	*Maximal*	*Mittel*	*Maximal*
U -233 -092	1.9E+06	3.8E+06	5.3E-03	1.1E-02
U -235 -092	2.0E+08	3.9E+08	2.5E+03	4.9E+03
Pu-238 -094	4.1E+13	8.1E+13	6.4E+01	1.3E+02
Pu-239 -094	4.1E+12	8.1E+12	1.8E+03	3.5E+03
Pu-241 -094	1.1E+14	2.2E+14	2.9E+01	5.8E+01
Am-242M-095	1.1E+11	2.3E+11	3.0E-01	5.9E-01
Cm-243 -096	1.0E+11	2.0E+11	5.4E-02	1.1E-01
Cm-245 -096	8.0E+09	1.6E+10	1.3E+00	2.5E+00
Total	**1.6E+14**	**3.1E+14**	**4.3E+03**	**8.6E+03**

Dosisleistungen an Gebindeoberfläche (0m)

Strahlungsart	*Dosisleistung [mSv/h]*	
	Mittel	*Maximal*
Gamma	1.1E+05	2.2E+05
Neutronen	3.0E+00	6.5E+00

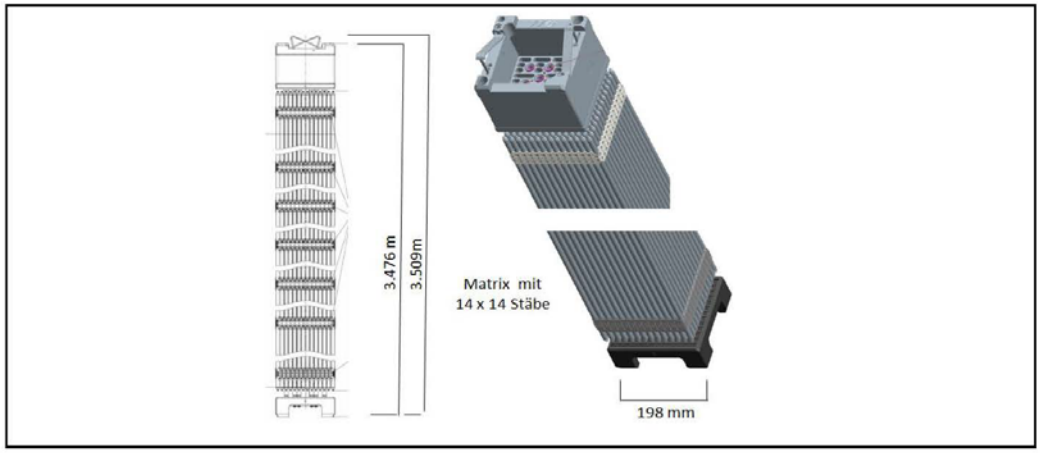
Mittlere Gesamtaktivität, Wärmeleistung, RTI (ab Referenzjahr)

Jahre	*Aktivität [Bq]*			*Wärmeleistung [W]*			*RTI*		
	∑α	*∑βγ*	*Total*	*∑α*	*∑βγ*	*Total*	*∑α*	*∑βγ*	*Total*
3	1.2E+14	1.5E+15	1.7E+15	1.0E+02	1.1E+02	2.1E+02	2.4E+11	1.5E+11	3.9E+11
10	1.1E+14	1.3E+15	1.4E+15	1.0E+02	9.2E+01	1.9E+02	2.4E+11	1.3E+11	3.6E+11
30	1.0E+14	7.9E+14	9.0E+14	9.3E+01	5.8E+01	1.5E+02	2.2E+11	7.8E+10	3.0E+11
100	8.2E+13	1.6E+14	2.4E+14	7.3E+01	1.1E+01	8.5E+01	1.8E+11	1.5E+10	1.9E+11
1000	2.4E+13	4.9E+12	2.8E+13	2.1E+01	4.3E-02	2.1E+01	5.2E+10	9.8E+06	5.2E+10
3000	1.0E+13	4.6E+12	1.5E+13	8.7E+00	3.8E-02	8.7E+00	2.5E+10	1.0E+07	2.5E+10
10000	6.1E+12	3.7E+12	9.8E+12	5.1E+00	2.6E-02	5.2E+00	1.5E+10	2.3E+07	1.5E+10
30000	2.3E+12	2.2E+12	4.5E+12	1.9E+00	1.7E-02	2.0E+00	5.6E+09	6.2E+07	5.7E+09
100000	4.9E+11	6.3E+11	1.1E+12	4.2E-01	1.9E-02	4.4E-01	1.1E+09	1.5E+08	1.2E+09
300000	2.8E+11	3.1E+11	5.9E+11	2.5E-01	1.9E-02	2.7E-01	5.3E+08	1.7E+08	6.9E+08
1000000	1.6E+11	1.5E+11	3.2E+11	1.6E-01	8.9E-03	1.6E-01	2.3E+08	6.5E+07	3.0E+08

Appendix 2: Swiss KK Beznau (KKB) detailed, representative PWR 14x14 MOX spent fuel characteristics and isotopes

MIRAM 14 - Basisszenarium MIRAM 14	Referenzjahr: 2075
BE-B-MOX-U-HAA - abgebrannte MOX-Brennelemente KKB unverpackt	

SCHEMATISCHE DARSTELLUNG DES REPRÄSENTATIVEN GEBINDES



AUFBAU DER ABFALLSORTE AUS ABFALLGEBINDETYPEN (AGT)

Anzahl			*pro Gebinde*	
Gebinde	*AGT Code*	*Bezeichnung*	*m^3*	*kg*
56	**J-B-950501**	@ BE KKB (MOX): 0.71/2.73% 33.8 GWd/t	1.38E-01	4.81E+02
36	**J-B-950502**	@ BE KKB (MOX): 0.26/3.66% 36.4 GWd/t	1.38E-01	4.81E+02
48	**J-B-950503**	@ BE KKB (MOX): 0.26/3.69% 43.0 GWd/t	1.38E-01	4.81E+02
52	**J-B-950504**	@ BE KKB (MOX): 0.27/4.81% 53.9 GWd/t	1.38E-01	4.81E+02
40	**J-B-950505**	@ BE KKB (MOX): 0.27/4.86% 55.0 GWd/t	1.38E-01	4.81E+02
232	**BE-B-MOX-U-HAA**	abgebrannte MOX-Brennelemente KKB unverpackt	**1.38E-01**	**4.81E+02**

BE-B-MOX-U-HAA - abgebrannte MOX-Brennelemente KKB unverpackt

MATERIALKENNDATEN FÜR DAS REPRÄSENTATIVE GEBINDE

Materialinventar und Herkunft

Material	*Masse [kg]*	*	*Herkunft und Masse [kg]*				
			C Abfallprodukt	*D* Behälter	*E* Füllmaterial	*F* Einbauten	*G* Leerraum
Inconel 718	**7.36E-01**	M				7.36E-01	
Luft	**9.59E-02**	A					9.59E-02
Plutonium(IV)oxid PuO2	**1.47E+01**	A	1.47E+01				
Spalt/Akt.produktoxide BS	**1.78E+01**	A	1.78E+01				
Stahl 1.4541	**1.22E+01**	M				1.22E+01	
Stahl 1.4571	**1.97E+00**	M				1.97E+00	
Uran(IV)oxid UO2	**3.31E+02**	A	3.31E+02				
Zircaloy-4	**1.03E+02**	M		8.74E+01		1.60E+01	
Total	**4.81E+02**		**3.63E+02**	**8.74E+01**	**0.00E+00**	**3.10E+01**	**9.59E-02**
* total anorganisch/nichtmetallisch	**3.63E+02**	A	3.63E+02				9.59E-02
metallisch	**1.18E+02**	M		8.74E+01		3.10E+01	

Oberflächen/Massen-Verhältnisse von Metallen

Material	*m^2*	*kg*	*m^2/kg*
Inconel 718	1.77E-01	7.36E-01	2.41E-01
Stahl 1.4541	3.12E-01	1.22E+01	2.55E-02
Stahl 1.4571	5.03E-01	1.97E+00	2.55E-01
Zircaloy-4	4.25E+01	1.03E+02	4.11E-01

BE-B-MOX-U-HAA - abgebrannte MOX-Brennelemente KKB unverpackt

RADIOLOGISCHE KENNDATEN FÜR DAS REPRÄSENTATIVE GEBINDE

Nuklidinventar

Nuklid	Aktivität [Bq]		Nuklid	Aktivität [Bq]		Nuklid	Aktivität [Bq]		Nuklid	Aktivität [Bq]	
	Mittel	Maximal		Mittel	Maximal		Mittel	Maximal		Mittel	Maximal
H-3	1.8E+11	3.7E+11	Cs-134	4.0E+05	7.9E+05	Pb-214	1.2E+05	2.4E+05	U-235	4.1E+07	8.1E+07
Be-10	1.1E+05	2.1E+05	Cs-135	1.4E+10	2.9E+10	Bi-208	1.4E+03	2.8E+03	U-235M	1.2E+13	2.3E+13
C-14	9.1E+09	1.8E+10	Cs-137	3.2E+14	6.3E+14	Bi-210	4.9E+04	9.7E+04	U-236	3.8E+08	7.7E+08
Si-32	3.0E+02	5.9E+02	Ba-133	1.3E+06	2.7E+06	Bi-210M	3.2E+02	6.3E+02	U-237	6.6E+09	1.3E+10
P-32	3.0E+02	5.9E+02	Ba-137M	3.0E+14	6.0E+14	Bi-211	5.5E+04	1.1E+05	U-238	3.6E+09	7.2E+09
Cl-36	1.7E+08	3.4E+08	La-137	1.1E+05	2.2E+05	Bi-212	1.6E+08	3.1E+08	U-240	2.7E+04	5.4E+04
Ar-39	2.4E+09	4.8E+09	La-138	3.7E+00	7.4E+00	Bi-213	3.5E+03	7.1E+03	Np-236	7.9E+04	1.6E+05
Ar-42	2.0E+00	4.0E+00	Nd-144	1.8E+01	3.5E+01	Bi-214	1.2E+05	2.4E+05	Np-237	6.8E+09	1.4E+10
K-40	4.3E+03	8.6E+03	Pm-145	3.0E+07	6.0E+07	Po-210	4.7E+04	9.5E+04	Np-238	7.0E+09	1.4E+10
K-42	2.0E+00	4.0E+00	Pm-146	1.2E+07	2.4E+07	Po-211	1.5E+02	3.0E+02	Np-239	2.9E+12	5.8E+12
Ca-41	2.3E+07	4.6E+07	Pm-147	3.3E+07	6.5E+07	Po-212	1.0E+08	2.0E+08	Np-240	3.0E+01	5.9E+01
Fe-55	1.3E+06	2.6E+06	Sm-146	5.4E+03	1.1E+04	Po-213	3.5E+03	6.9E+03	Np-240M	2.7E+04	5.4E+04
Co-60	2.1E+09	4.3E+09	Sm-147	8.4E+04	1.7E+05	Po-214	1.2E+05	2.4E+05	Pu-236	1.0E+04	2.1E+04
Ni-59	2.4E+10	4.8E+10	Sm-151	7.0E+12	1.4E+13	Po-215	5.5E+04	1.1E+05	Pu-238	9.9E+13	2.0E+14
Ni-63	1.9E+12	3.8E+12	Eu-150	9.6E+05	1.9E+06	Po-216	1.6E+08	3.1E+08	Pu-239	1.2E+13	2.3E+13
Se-79	9.2E+08	1.8E+09	Eu-152	1.0E+10	2.1E+10	Po-218	1.2E+05	2.4E+05	Pu-240	3.6E+13	7.2E+13
Kr-81	2.9E+05	5.8E+05	Eu-154	7.9E+11	1.6E+12	At-217	3.5E+03	7.1E+03	Pu-241	2.7E+14	5.4E+14
Kr-85	7.5E+11	1.5E+12	Eu-155	2.7E+09	5.4E+09	At-218	2.4E+01	4.8E+01	Pu-242	2.0E+11	3.9E+11
Rb-87	1.7E+05	3.4E+05	Tb-157	1.4E+07	2.8E+07	Rn-219	5.5E+04	1.1E+05	Pu-243	1.7E+05	3.3E+05
Sr-90	1.0E+14	2.0E+14	Ho-163	1.3E+06	2.5E+06	Rn-220	1.6E+08	3.1E+08	Pu-244	2.7E+04	5.4E+04
Y-90	1.0E+14	2.0E+14	Ho-166M	1.8E+08	3.6E+08	Rn-222	1.2E+05	2.4E+05	Am-241	2.8E+14	5.5E+14
Zr-93	1.5E+10	3.0E+10	Tm-171	3.3E+00	6.6E+00	Fr-221	3.5E+03	7.1E+03	Am-242	1.5E+12	2.9E+12
Nb-91	8.8E+02	1.8E+03	Lu-176	3.2E+00	6.5E+00	Fr-223	7.6E+02	1.5E+03	Am-242M	1.5E+12	2.9E+12
Nb-92	6.4E+01	1.3E+02	Hf-182	1.2E+02	2.4E+02	Ra-223	5.5E+04	1.1E+05	Am-243	2.9E+12	5.8E+12
Nb-93M	1.4E+10	2.7E+10	Ta-182	1.2E+02	2.4E+02	Ra-224	1.6E+08	3.1E+08	Cm-242	1.2E+12	2.4E+12
Nb-94	7.7E+09	1.5E+10	Re-187	1.1E+02	2.2E+02	Ra-225	3.5E+03	7.1E+03	Cm-243	5.3E+11	1.1E+12
Mo-93	1.5E+07	3.1E+07	Os-194	4.6E+02	9.1E+02	Ra-226	1.2E+05	2.4E+05	Cm-244	4.9E+13	9.7E+13
Tc-97	1.5E+03	3.1E+03	Ir-192	1.9E+06	3.9E+06	Ac-225	3.5E+03	7.1E+03	Cm-245	1.7E+11	3.4E+11
Tc-98	1.5E+05	3.0E+05	Ir-192M	1.9E+06	3.9E+06	Ac-227	5.5E+04	1.1E+05	Cm-246	2.8E+10	5.5E+10
Tc-99	2.0E+11	4.1E+11	Ir-194	4.6E+02	9.1E+02	Th-227	5.4E+04	1.1E+05	Cm-247	1.7E+05	3.3E+05
Pd-107	4.1E+09	8.1E+09	Pt-193	1.0E+09	2.0E+09	Th-228	1.6E+08	3.1E+08	Cm-248	5.3E+05	1.1E+06
Ag-108	1.5E+08	3.1E+08	Tl-204	2.3E+04	4.6E+04	Th-229	3.5E+03	7.1E+03	Cf-249	1.1E+07	2.2E+07
Ag-108M	1.8E+09	3.5E+09	Tl-206	3.2E+02	6.3E+02	Th-230	1.1E+07	2.1E+07	Cf-250	1.7E+06	3.5E+06
Cd-113M	3.0E+10	6.0E+10	Tl-207	5.5E+04	1.1E+05	Th-231	4.1E+07	8.1E+07	Cf-251	6.8E+05	1.4E+06
Sn-121	2.3E+10	4.7E+10	Tl-208	5.6E+07	1.1E+08	Th-234	3.6E+09	7.2E+09	Cf-252	5.3E+00	1.1E+01
Sn-121M	3.0E+10	6.0E+10	Tl-209	7.6E+01	1.5E+02	Pa-231	8.1E+04	1.6E+05	Cf-254	1.1E+02	2.2E+02
Sn-126	1.4E+10	2.8E+10	Tl-210	2.5E+01	5.1E+01	Pa-232	1.3E+02	2.5E+02			
Sb-125	6.7E+06	1.3E+07	Pb-205	6.3E+01	1.3E+02	Pa-233	6.8E+09	1.4E+10			
Sb-126	2.0E+09	3.9E+09	Pb-209	3.5E+03	7.1E+03	Pa-234	3.6E+09	7.2E+09			
Sb-126M	1.4E+10	2.8E+10	Pb-210	4.9E+04	9.7E+04	U-232	1.5E+08	3.0E+08	Summe α	4.8E+14	9.6E+14
Te-125M	1.6E+06	3.2E+06	Pb-211	5.5E+04	1.1E+05	U-233	1.3E+06	2.5E+06	Summe βγ	1.1E+15	2.2E+15
I-129	6.2E+08	1.2E+09	Pb-212	1.6E+08	3.1E+08	U-234	2.8E+10	5.6E+10	**Total**	**1.6E+15**	**3.2E+15**

MIRAM 14 - Basisszenarium MIRAM 14	Referenzjahr: 2075
BE-B-MOX-U-HAA - abgebrannte MOX-Brennelemente KKB unverpackt	

Spaltstoffinventar

Nuklid	*Aktivität [Bq]*		*Masse [g]*	
	Mittel	*Maximal*	*Mittel*	*Maximal*
U -233 -092	1.3E+06	2.5E+06	3.5E-03	7.0E-03
U -235 -092	4.1E+07	8.1E+07	5.1E+02	1.0E+03
Pu-238 -094	9.9E+13	2.0E+14	1.6E+02	3.1E+02
Pu-239 -094	1.2E+13	2.3E+13	5.0E+03	1.0E+04
Pu-241 -094	2.7E+14	5.4E+14	7.0E+01	1.4E+02
Am-242M-095	1.5E+12	2.9E+12	3.8E+00	7.6E+00
Cm-243 -096	5.3E+11	1.1E+12	2.8E-01	5.6E-01
Cm-245 -096	1.7E+11	3.4E+11	2.7E+01	5.4E+01
Total	**3.8E+14**	**7.7E+14**	**5.8E+03**	**1.2E+04**

Dosisleistungen an Gebindeoberfläche (0m)

Strahlungsart	*Dosisleistung [mSv/h]*	
	Mittel	*Maximal*
Gamma	7.5E+04	1.5E+05
Neutronen	2.6E+01	5.1E+01

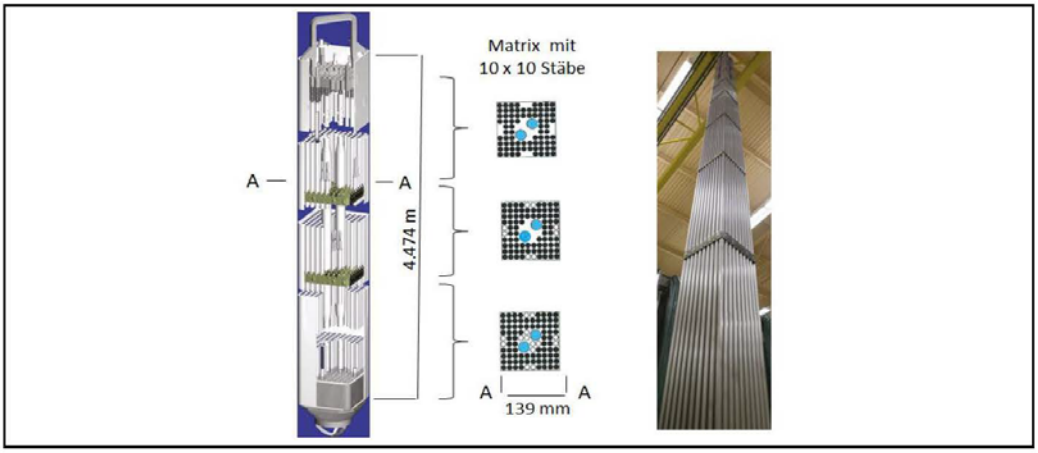
Mittlere Gesamtaktivität, Wärmeleistung, RTI (ab Referenzjahr)

Jahre	*Aktivität [Bq]*			*Wärmeleistung [W]*			*RTI*		
	∑α	*∑βγ*	*Total*	*∑α*	*∑βγ*	*Total*	*∑α*	*∑βγ*	*Total*
3	4.7E+14	1.0E+15	1.5E+15	4.2E+02	5.6E+01	4.8E+02	9.5E+11	8.1E+10	1.0E+12
10	4.5E+14	8.4E+14	1.3E+15	4.1E+02	4.7E+01	4.5E+02	9.3E+11	6.7E+10	1.0E+12
30	4.2E+14	4.9E+14	9.1E+14	3.7E+02	3.0E+01	4.0E+02	8.7E+11	4.1E+10	9.1E+11
100	3.4E+14	1.0E+14	4.4E+14	3.0E+02	6.0E+00	3.1E+02	7.2E+11	8.7E+09	7.3E+11
1000	1.0E+14	1.5E+13	1.2E+14	9.2E+01	1.9E-01	9.2E+01	2.3E+11	5.4E+07	2.3E+11
3000	4.2E+13	1.4E+13	5.6E+13	3.6E+01	1.6E-01	3.6E+01	1.0E+11	3.3E+07	1.0E+11
10000	2.3E+13	1.1E+13	3.4E+13	2.0E+01	9.3E-02	2.0E+01	5.8E+10	4.7E+07	5.8E+10
30000	7.5E+12	6.0E+12	1.3E+13	6.3E+00	3.4E-02	6.4E+00	1.8E+10	1.1E+08	1.9E+10
100000	1.3E+12	1.2E+12	2.6E+12	1.2E+00	3.2E-02	1.2E+00	3.0E+09	2.5E+08	3.2E+09
300000	7.0E+11	4.8E+11	1.2E+12	6.4E-01	3.4E-02	6.8E-01	1.2E+09	3.0E+08	1.5E+09
1000000	4.6E+11	2.8E+11	7.4E+11	4.4E-01	1.8E-02	4.5E-01	5.7E+08	1.2E+08	6.9E+08

Appendix 3: Swiss KK Mühleberg (KKM) detailed, representative BWR 10x10 UO_2 spent fuel characteristics and isotopes

MIRAM 14 - Basisszenarium MIRAM 14	Referenzjahr: 2075
BE-M-UO2-U-HAA - abgebrannte UO2-Brennelemente KKM unverpackt	

SCHEMATISCHE DARSTELLUNG DES REPRÄSENTATIVEN GEBINDES



AUFBAU DER ABFALLSORTE AUS ABFALLGEBINDETYPEN (AGT)

Anzahl			*pro Gebinde*	
Gebinde	*AGT Code*	*Bezeichnung*	*m^3*	*kg*
185	**J-M-950001**	@ BE KKM (UO2): 3.13% 40.3 GWd/t	8.66E-02	2.71E+02
242	**J-M-950002**	@ BE KKM (UO2): 3.67% 48.5 GWd/t	8.66E-02	2.71E+02
260	**J-M-950003**	@ BE KKM (UO2): 4.08% 52.1 GWd/t	8.66E-02	2.71E+02
300	**J-M-950004**	@ BE KKM (UO2): 4.20% 55.0 GWd/t	8.66E-02	2.71E+02
200	**J-M-950005**	@ BE KKM (UO2): 4.20% 30.8 GWd/t	8.66E-02	2.71E+02
1187	**BE-M-UO2-U-HAA**	abgebrannte UO2-Brennelemente KKM unverpackt	**8.66E-02**	**2.71E+02**

BE-M-UO2-U-HAA - abgebrannte UO2-Brennelemente KKM unverpackt

MATERIALKENNDATEN FÜR DAS REPRÄSENTATIVE GEBINDE

Materialinventar und Herkunft

Material	*Masse [kg]*	*	*Herkunft und Masse [kg]*				
			C	*D*	*E*	*F*	*G*
			Abfallprodukt	*Behälter*	*Füllmaterial*	*Einbauten*	*Leerraum*
Inconel 600	**9.03E-01**	M				9.03E-01	
Luft	**6.32E-02**	A					6.32E-02
Plutonium(IV)oxid PuO2	**1.65E+00**	A	1.65E+00				
Spalt/Akt.produktoxide BS	**9.97E+00**	A	9.97E+00				
Stahl 12.03	**1.00E+01**	M				1.00E+01	
Uran(IV)oxid UO2	**1.85E+02**	A	1.85E+02				
Zircaloy-2	**6.39E+01**	M		4.75E+01		1.64E+01	
Total	**2.71E+02**		**1.96E+02**	**4.75E+01**	**0.00E+00**	**2.74E+01**	**6.32E-02**
* total anorganisch/nichtmetallisch	**1.96E+02**	A	1.96E+02				6.32E-02
metallisch	**7.48E+01**	M		4.75E+01		2.74E+01	

Oberflächen/Massen-Verhältnisse von Metallen

Material	*m^2*	*kg*	*m^2/kg*
Inconel 600	2.17E-01	9.03E-01	2.41E-01
Stahl 12.03	2.55E+00	1.00E+01	2.55E-01
Zircaloy-2	2.49E+01	6.39E+01	3.90E-01

BE-M-UO2-U-HAA - abgebrannte UO2-Brennelemente KKM unverpackt

RADIOLOGISCHE KENNDATEN FÜR DAS REPRÄSENTATIVE GEBINDE

Nuklidinventar

Nuklid	Aktivität [Bq]		Nuklid	Aktivität [Bq]		Nuklid	Aktivität [Bq]		Nuklid	Aktivität [Bq]	
	Mittel	Maximal		Mittel	Maximal		Mittel	Maximal		Mittel	Maximal
H-3	1.8E+11	3.5E+11	Cs-135	3.6E+09	7.1E+09	Bi-210M	1.9E+02	3.8E+02	U-235M	1.5E+12	2.9E+12
Be-10	1.9E+06	3.8E+06	Cs-137	2.0E+14	4.1E+14	Bi-211	1.7E+05	3.4E+05	U-236	2.3E+09	4.5E+09
C-14	1.3E+10	2.7E+10	Ba-133	7.8E+07	1.6E+08	Bi-212	9.3E+07	1.9E+08	U-237	6.7E+08	1.3E+09
Si-32	2.3E+04	4.7E+04	Ba-137M	1.9E+14	3.9E+14	Bi-213	4.2E+03	8.5E+03	U-238	2.0E+09	4.0E+09
P-32	2.3E+04	4.7E+04	La-137	8.2E+04	1.6E+05	Bi-214	8.3E+04	1.7E+05	U-240	3.5E+03	6.9E+03
Cl-36	2.6E+08	5.3E+08	La-138	1.6E+00	3.1E+00	Po-210	3.7E+04	7.3E+04	Np-236	1.1E+04	2.2E+04
Ar-39	6.3E+08	1.3E+09	Nd-144	1.6E+01	3.2E+01	Po-211	4.6E+02	9.3E+02	Np-237	2.6E+09	5.2E+09
Ar-42	1.0E+02	2.0E+02	Pm-145	1.0E+07	2.1E+07	Po-212	6.0E+07	1.2E+08	Np-238	1.3E+08	2.5E+08
K-40	4.9E+03	9.8E+03	Pm-146	4.9E+05	9.8E+05	Po-213	4.2E+03	8.3E+03	Np-239	2.5E+11	4.9E+11
K-42	1.0E+02	2.0E+02	Pm-147	1.2E+08	2.4E+08	Po-214	8.3E+04	1.7E+05	Np-240	3.8E+00	7.6E+00
Ca-41	3.1E+07	6.1E+07	Sm-146	1.8E+03	3.5E+03	Po-215	1.7E+05	3.4E+05	Np-240M	3.5E+03	6.9E+03
Fe-55	5.9E+06	1.2E+07	Sm-147	4.8E+04	9.6E+04	Po-216	9.3E+07	1.9E+08	Pu-236	2.0E+03	3.9E+03
Co-60	4.3E+09	8.6E+09	Sm-151	1.2E+12	2.4E+12	Po-218	8.3E+04	1.7E+05	Pu-238	1.4E+13	2.8E+13
Ni-59	1.4E+10	2.8E+10	Eu-152	1.2E+09	2.4E+09	At-217	4.2E+03	8.5E+03	Pu-239	1.5E+12	2.9E+12
Ni-63	1.2E+12	2.4E+12	Eu-154	2.5E+11	5.1E+11	At-218	1.7E+01	3.3E+01	Pu-240	3.7E+12	7.5E+12
Se-79	6.5E+08	1.3E+09	Eu-155	2.3E+09	4.7E+09	Rn-219	1.7E+05	3.4E+05	Pu-241	2.7E+13	5.5E+13
Kr-81	1.9E+05	3.8E+05	Tb-157	1.2E+06	2.3E+06	Rn-220	9.3E+07	1.9E+08	Pu-242	2.2E+10	4.4E+10
Kr-85	1.3E+12	2.5E+12	Tb-158	1.7E+05	3.4E+05	Rn-222	8.3E+04	1.7E+05	Pu-243	2.7E+03	5.5E+03
Rb-87	1.9E+05	3.8E+05	Ho-163	2.3E+05	4.6E+05	Fr-221	4.2E+03	8.5E+03	Pu-244	3.5E+03	6.9E+03
Sr-90	1.3E+14	2.7E+14	Ho-166M	7.8E+07	1.6E+08	Fr-223	2.3E+03	4.6E+03	Pu-246	1.9E+00	3.9E+00
Y-90	1.3E+14	2.7E+14	Tm-171	1.8E+01	3.7E+01	Ra-223	1.7E+05	3.4E+05	Am-241	2.0E+13	4.0E+13
Zr-93	1.8E+10	3.6E+10	Hf-182	2.7E+02	5.5E+02	Ra-224	9.3E+07	1.9E+08	Am-242	2.6E+10	5.2E+10
Nb-91	3.2E+02	6.3E+02	Ta-182	2.7E+02	5.5E+02	Ra-225	4.2E+03	8.5E+03	Am-242M	2.6E+10	5.2E+10
Nb-92	2.3E+01	4.7E+01	Re-187	5.2E+01	1.0E+02	Ra-226	8.3E+04	1.7E+05	Am-243	2.5E+11	4.9E+11
Nb-93M	1.6E+10	3.2E+10	Os-194	7.5E+01	1.5E+02	Ra-228	6.8E+00	1.4E+01	Am-246	1.9E+00	3.9E+00
Nb-94	1.4E+09	2.9E+09	Ir-192	1.6E+04	3.2E+04	Ac-225	4.2E+03	8.5E+03	Cm-242	2.2E+10	4.3E+10
Mo-93	6.0E+06	1.2E+07	Ir-192M	1.6E+04	3.2E+04	Ac-227	1.7E+05	3.4E+05	Cm-243	3.0E+10	6.0E+10
Tc-97	5.5E+02	1.1E+03	Ir-194	7.5E+01	1.5E+02	Ac-228	6.8E+00	1.4E+01	Cm-244	2.8E+12	5.6E+12
Tc-98	5.3E+04	1.1E+05	Pt-193	5.0E+08	1.0E+09	Th-227	1.7E+05	3.3E+05	Cm-245	2.1E+09	4.1E+09
Tc-99	1.2E+11	2.4E+11	Tl-204	3.7E+04	7.5E+04	Th-228	9.3E+07	1.9E+08	Cm-246	8.6E+08	1.7E+09
Pd-107	1.0E+09	2.0E+09	Tl-206	1.9E+02	3.8E+02	Th-229	4.3E+03	8.5E+03	Cm-247	2.7E+03	5.5E+03
Ag-108	7.6E+07	1.5E+08	Tl-207	1.7E+05	3.4E+05	Th-230	5.8E+06	1.2E+07	Cm-248	1.1E+04	2.2E+04
Ag-108M	8.7E+08	1.7E+09	Tl-208	3.3E+07	6.7E+07	Th-231	6.9E+07	1.4E+08	Cm-250	3.0E+00	6.1E+00
Cd-113M	1.0E+08	2.0E+08	Tl-209	9.2E+01	1.8E+02	Th-232	7.7E+00	1.5E+01	Bk-250	1.1E+00	2.2E+00
Sn-121	4.3E+10	8.6E+10	Tl-210	1.7E+01	3.5E+01	Th-234	2.0E+09	4.0E+09	Cf-249	8.9E+04	1.8E+05
Sn-121M	5.6E+10	1.1E+11	Pb-205	4.5E+01	9.0E+01	Pa-231	2.2E+05	4.5E+05	Cf-250	2.6E+04	5.2E+04
Sn-126	4.7E+09	9.4E+09	Pb-209	4.2E+03	8.5E+03	Pa-232	1.8E+01	3.5E+01	Cf-251	4.8E+03	9.6E+03
Sb-125	1.6E+07	3.2E+07	Pb-210	3.7E+04	7.5E+04	Pa-233	2.6E+09	5.2E+09	Cf-254	1.0E+03	2.1E+03
Sb-126	6.6E+08	1.3E+09	Pb-211	1.7E+05	3.4E+05	Pa-234	2.0E+09	4.0E+09			
Sb-126M	4.7E+09	9.4E+09	Pb-212	9.3E+07	1.9E+08	U-232	9.0E+07	1.8E+08			
Te-125M	3.9E+06	7.8E+06	Pb-214	8.3E+04	1.7E+05	U-233	8.0E+05	1.6E+06	Summe α	4.3E+13	8.5E+13
I-129	2.3E+08	4.7E+08	Bi-208	5.5E+02	1.1E+03	U-234	1.1E+10	2.1E+10	Summe βγ	7.0E+14	1.4E+15
Cs-134	1.5E+06	2.9E+06	Bi-210	3.7E+04	7.5E+04	U-235	6.9E+07	1.4E+08	**Total**	**7.4E+14**	**1.5E+15**

MIRAM 14 - Basisszenarium MIRAM 14	Referenzjahr: 2075
BE-M-UO2-U-HAA - abgebrannte UO2-Brennelemente KKM unverpackt	

Spaltstoffinventar

Nuklid	*Aktivität [Bq]*		*Masse [g]*	
	Mittel	*Maximal*	*Mittel*	*Maximal*
U -233 -092	8.0E+05	1.6E+06	2.2E-03	4.5E-03
U -235 -092	6.9E+07	1.4E+08	8.6E+02	1.7E+03
Pu-238 -094	1.4E+13	2.8E+13	2.2E+01	4.4E+01
Pu-239 -094	1.5E+12	2.9E+12	6.4E+02	1.3E+03
Pu-241 -094	2.7E+13	5.5E+13	7.2E+00	1.4E+01
Am-242M-095	2.6E+10	5.2E+10	6.8E-02	1.4E-01
Cm-243 -096	3.0E+10	6.0E+10	1.6E-02	3.2E-02
Cm-245 -096	2.1E+09	4.1E+09	3.3E-01	6.5E-01
Total	**4.3E+13**	**8.6E+13**	**1.5E+03**	**3.1E+03**

Dosisleistungen an Gebindeoberfläche (0m)

Strahlungsart	*Dosisleistung [mSv/h]*	
	Mittel	*Maximal*
Gamma	5.3E+04	1.1E+05
Neutronen	1.5E+00	3.0E+00

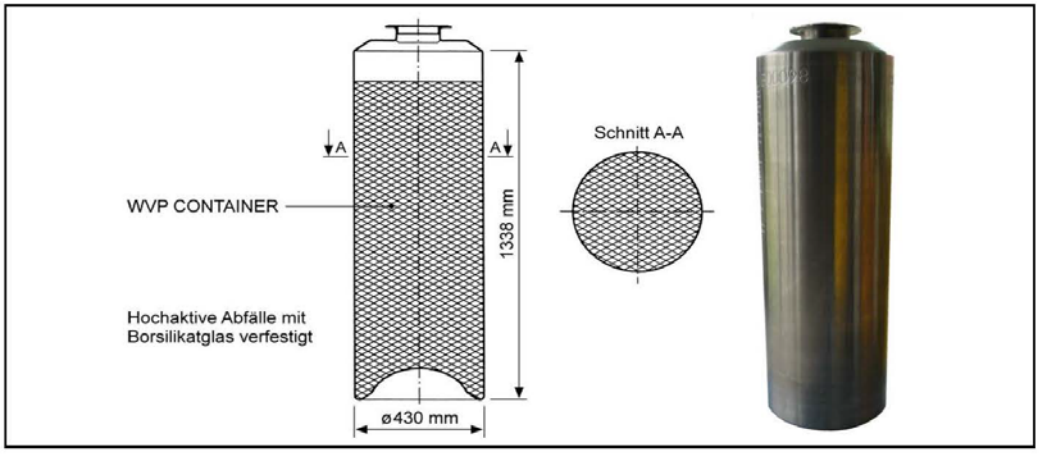
Mittlere Gesamtaktivität, Wärmeleistung, RTI (ab Referenzjahr)

Jahre	*Aktivität [Bq]*			*Wärmeleistung [W]*			*RTI*		
	∑α	*∑βγ*	*Total*	*∑α*	*∑βγ*	*Total*	*∑α*	*∑βγ*	*Total*
3	4.2E+13	6.5E+14	6.9E+14	3.8E+01	4.8E+01	8.5E+01	8.9E+10	6.4E+10	1.5E+11
10	4.1E+13	5.5E+14	5.9E+14	3.6E+01	4.0E+01	7.7E+01	8.6E+10	5.4E+10	1.4E+11
30	3.7E+13	3.4E+14	3.8E+14	3.3E+01	2.5E+01	5.9E+01	8.0E+10	3.4E+10	1.1E+11
100	3.0E+13	6.7E+13	9.7E+13	2.7E+01	4.9E+00	3.2E+01	6.4E+10	6.4E+09	7.1E+10
1000	9.3E+12	1.9E+12	1.1E+13	8.1E+00	2.0E-02	8.1E+00	2.1E+10	4.3E+06	2.1E+10
3000	4.5E+12	1.8E+12	6.2E+12	3.8E+00	1.8E-02	3.8E+00	1.1E+10	4.7E+06	1.1E+10
10000	2.6E+12	1.4E+12	4.0E+12	2.2E+00	1.2E-02	2.2E+00	6.4E+09	1.0E+07	6.4E+09
30000	9.0E+11	8.7E+11	1.8E+12	7.6E-01	7.8E-03	7.6E-01	2.2E+09	2.6E+07	2.2E+09
100000	1.9E+11	2.8E+11	4.7E+11	1.7E-01	8.7E-03	1.8E-01	4.3E+08	6.1E+07	4.9E+08
300000	1.2E+11	1.5E+11	2.6E+11	1.1E-01	8.3E-03	1.1E-01	2.3E+08	7.0E+07	3.0E+08
1000000	6.8E+10	6.9E+10	1.4E+11	6.4E-02	4.0E-03	6.8E-02	1.0E+08	2.9E+07	1.3E+08

Appendix 4: Characteristics and isotopes of a representative "CSD-V" universal canister containing fission products (residues) from Swiss reprocessing campaigns

WA-F-KG-K1-HAA - verglaste Konzentrate HAA AREVA (CSD-V) in 180-l-Kokille

SCHEMATISCHE DARSTELLUNG DES REPRÄSENTATIVEN GEBINDES



AUFBAU DER ABFALLSORTE AUS ABFALLGEBINDETYPEN (AGT)

Anzahl			*pro Gebinde*	
Gebinde	*AGT Code*	*Bezeichnung*	*m^3*	*kg*
438	**J-Z-005000**	COGEMA VITRIFIED RESIDUE	1.82E-01	4.89E+02
438	**WA-F-KG-K1-HAA**	verglaste Konzentrate HAA AREVA (CSD-V) in 180-l-Kokille	**1.82E-01**	**4.89E+02**

WA-F-KG-K1-HAA - verglaste Konzentrate HAA AREVA (CSD-V) in 180-l-Kokille

MATERIALKENNDATEN FÜR DAS REPRÄSENTATIVE GEBINDE

Materialinventar und Herkunft

Material		*Masse [kg]*	*	*Herkunft und Masse [kg]* C Abfallprodukt	D Behälter	E Füllmaterial	F Einbauten	G Leerraum
::WA/HLW-Glasmatrix-::Glas		**3.33E+02**	A	3.33E+02				
::WA/HLW-Glasmatrix-::PuO2		**1.73E-02**	A	1.73E-02				
::WA/HLW-Glasmatrix-::SPOx		**6.08E+01**	A	6.08E+01				
::WA/HLW-Glasmatrix-::UO2		**1.80E+00**	A	1.80E+00				
CrNi-Stahl		**9.40E+01**	M		9.40E+01			
Luft		**2.19E-02**	A					2.19E-02
Total		**4.89E+02**		**3.95E+02**	**9.40E+01**	**0.00E+00**	**0.00E+00**	**2.19E-02**
* total	anorganisch/nichtmetallisch	**3.95E+02**	A	3.95E+02				2.19E-02
	metallisch	**9.40E+01**	M		9.40E+01			

Oberflächen/Massen-Verhältnisse von Metallen

Material	m^2	kg	m^2/kg
CrNi-Stahl	4.76E+00	9.40E+01	5.06E-02

WA-F-KG-K1-HAA - verglaste Konzentrate HAA AREVA (CSD-V) in 180-l-Kokille

RADIOLOGISCHE KENNDATEN FÜR DAS REPRÄSENTATIVE GEBINDE

Nuklidinventar

Nuklid	Aktivität [Bq]		Nuklid	Aktivität [Bq]		Nuklid	Aktivität [Bq]		Nuklid	Aktivität [Bq]	
	Mittel	Maximal		Mittel	Maximal		Mittel	Maximal		Mittel	Maximal
Be-10	3.0E+04	1.7E+05	Ba-137M	9.4E+14	1.1E+15	Po-216	6.1E+05	1.7E+06	U-234	1.4E+08	2.9E+08
C-14	2.9E+07	1.6E+08	Pm-146	7.2E+06	8.2E+06	Po-218	1.5E+05	7.4E+05	U-235	6.1E+05	2.4E+06
Ca-41	1.0E+06	5.8E+06	Pm-147	1.2E+07	1.3E+07	At-217	2.4E+04	7.3E+04	U-235M	2.2E+10	4.4E+10
Fe-55	7.0E+02	3.9E+03	Sm-146	1.4E+03	1.6E+03	At-218	3.0E+01	1.5E+02	U-236	9.8E+06	2.8E+07
Co-60	4.1E+07	2.8E+08	Sm-147	6.9E+04	7.7E+04	Rn-217	1.7E+00	5.1E+00	U-237	3.6E+06	8.1E+06
Ni-59	3.7E+08	2.0E+09	Sm-151	1.3E+13	1.7E+13	Rn-219	6.3E+05	8.2E+05	U-238	1.2E+07	4.6E+07
Ni-63	3.1E+10	1.7E+11	Eu-152	2.9E+09	3.1E+09	Rn-220	6.1E+05	1.7E+06	U-240	1.3E+01	3.3E+01
Se-79	1.9E+10	2.4E+10	Eu-154	4.7E+11	6.3E+11	Rn-222	1.5E+05	7.4E+05	Np-237	1.9E+10	4.4E+10
Sr-90	5.8E+14	6.9E+14	Eu-155	1.7E+09	4.5E+09	Fr-221	2.4E+04	7.3E+04	Np-238	6.3E+09	8.6E+09
Y-90	5.8E+14	6.9E+14	Ho-166M	2.6E+07	1.4E+08	Fr-223	8.6E+03	1.1E+04	Np-239	1.5E+12	2.1E+12
Zr-93	1.0E+11	1.2E+11	Tl-207	6.2E+05	8.1E+05	Ra-223	6.3E+05	8.2E+05	Np-240M	1.3E+01	3.3E+01
Nb-93M	9.2E+10	1.1E+11	Tl-208	2.2E+05	6.2E+05	Ra-224	6.1E+05	1.7E+06	Pu-238	6.9E+11	1.1E+12
Nb-94	1.0E+08	5.8E+08	Tl-209	5.3E+02	1.6E+03	Ra-225	2.4E+04	7.3E+04	Pu-239	2.2E+10	4.4E+10
Mo-93	4.8E+07	3.0E+08	Tl-210	3.1E+01	1.5E+02	Ra-226	1.5E+05	7.4E+05	Pu-240	4.2E+11	6.6E+11
Tc-99	9.5E+11	1.4E+12	Pb-209	2.4E+04	7.3E+04	Ra-228	5.2E+00	2.6E+01	Pu-241	1.5E+11	3.3E+11
Pd-107	6.5E+09	8.0E+09	Pb-210	9.1E+04	4.6E+05	Ac-225	2.4E+04	7.3E+04	Pu-242	1.6E+08	3.5E+08
Ag-108	6.8E+06	3.8E+07	Pb-211	6.3E+05	8.2E+05	Ac-227	6.3E+05	8.2E+05	Pu-244	1.3E+01	3.3E+01
Ag-108M	7.8E+07	4.3E+08	Pb-212	6.1E+05	1.7E+06	Ac-228	5.2E+00	2.6E+01	Am-241	7.8E+13	1.2E+14
Cd-113M	9.7E+09	5.4E+10	Pb-214	1.5E+05	7.4E+05	Th-227	6.2E+05	8.1E+05	Am-242	1.3E+12	1.8E+12
Sn-121	2.6E+09	3.9E+11	Bi-210	9.1E+04	4.5E+05	Th-228	6.0E+05	1.7E+06	Am-242M	1.3E+12	1.8E+12
Sn-121M	3.3E+09	5.0E+11	Bi-211	6.3E+05	8.2E+05	Th-229	2.4E+04	7.3E+04	Am-243	1.5E+12	2.1E+12
Sn-126	3.7E+10	4.5E+10	Bi-212	6.1E+05	1.7E+06	Th-230	4.6E+06	2.3E+07	Cm-242	1.1E+12	1.5E+12
Sb-125	4.4E+05	1.4E+06	Bi-213	2.4E+04	7.3E+04	Th-231	6.1E+05	2.4E+06	Cm-243	2.2E+11	3.8E+11
Sb-126	5.1E+09	6.3E+09	Bi-214	1.5E+05	7.4E+05	Th-232	5.2E+00	2.6E+01	Cm-244	8.5E+12	1.3E+13
Sb-126M	3.7E+10	4.5E+10	Po-210	9.0E+04	4.5E+05	Th-234	1.2E+07	4.6E+07	Cm-245	1.7E+10	3.5E+10
Te-125M	1.1E+05	3.4E+05	Po-211	1.7E+03	2.3E+03	Pa-231	6.8E+05	8.2E+05	Cm-246	2.7E+09	1.3E+10
I-129	3.6E+07	2.5E+08	Po-212	3.9E+05	1.1E+06	Pa-233	1.9E+10	4.4E+10	Cm-248	7.1E+03	3.4E+04
Cs-134	1.8E+04	3.4E+04	Po-213	2.4E+04	7.1E+04	Pa-234	1.2E+07	4.6E+07	Summe α	9.0E+13	1.4E+14
Cs-135	2.9E+10	3.5E+10	Po-214	1.5E+05	7.4E+05	U-232	5.9E+05	1.7E+06	Summe βγ	3.1E+15	3.6E+15
Cs-137	9.9E+14	1.1E+15	Po-215	6.3E+05	8.2E+05	U-233	5.9E+06	1.4E+07	**Total**	**3.2E+15**	**3.7E+15**

WA-F-KG-K1-HAA - verglaste Konzentrate HAA AREVA (CSD-V) in 180-l-Kokille

Spaltstoffinventar

Nuklid	*Aktivität [Bq]*		*Masse [g]*	
	Mittel	*Maximal*	*Mittel*	*Maximal*
U -233 -092	5.9E+06	1.4E+07	1.6E-02	3.9E-02
U -235 -092	6.1E+05	2.4E+06	7.6E+00	3.0E+01
Pu-238 -094	6.9E+11	1.1E+12	1.1E+00	1.7E+00
Pu-239 -094	2.2E+10	4.4E+10	9.4E+00	1.9E+01
Pu-241 -094	1.5E+11	3.3E+11	3.9E-02	8.6E-02
Am-242M-095	1.3E+12	1.8E+12	3.4E+00	4.6E+00
Cm-243 -096	2.2E+11	3.8E+11	1.2E-01	2.1E-01
Cm-245 -096	1.7E+10	3.5E+10	2.6E+00	5.6E+00
Total	**2.4E+12**	**3.6E+12**	**2.4E+01**	**6.1E+01**

Dosisleistungen an Gebindeoberfläche (0m)

Strahlungsart	*Dosisleistung [mSv/h]*	
	Mittel	*Maximal*
Gamma	3.0E+05	3.4E+05
Neutronen	1.7E+01	2.5E+01

Mittlere Gesamtaktivität, Wärmeleistung, RTI (ab Referenzjahr)

Jahre	*Aktivität [Bq]*			*Wärmeleistung [W]*			*RTI*		
	∑α	*∑βγ*	*Total*	*∑α*	*∑βγ*	*Total*	*∑α*	*∑βγ*	*Total*
3	8.9E+13	2.9E+15	3.0E+15	8.0E+01	2.2E+02	3.0E+02	1.7E+11	2.9E+11	4.6E+11
10	8.6E+13	2.5E+15	2.5E+15	7.8E+01	1.8E+02	2.6E+02	1.7E+11	2.4E+11	4.1E+11
30	8.0E+13	1.5E+15	1.6E+15	7.3E+01	1.2E+02	1.9E+02	1.6E+11	1.5E+11	3.1E+11
100	7.0E+13	3.1E+14	3.8E+14	6.3E+01	2.2E+01	8.5E+01	1.4E+11	3.0E+10	1.7E+11
1000	1.8E+13	2.8E+12	2.0E+13	1.6E+01	1.2E-01	1.6E+01	3.5E+10	4.1E+07	3.5E+10
3000	2.3E+12	2.6E+12	4.9E+12	2.0E+00	1.1E-01	2.1E+00	4.8E+09	2.1E+07	4.9E+09
10000	1.1E+12	2.2E+12	3.2E+12	9.2E-01	7.1E-02	9.9E-01	2.3E+09	1.7E+07	2.3E+09
30000	4.2E+11	1.6E+12	2.0E+12	3.6E-01	3.7E-02	4.0E-01	9.4E+08	1.5E+07	9.5E+08
100000	1.4E+11	1.1E+12	1.2E+12	1.3E-01	2.7E-02	1.6E-01	2.1E+08	2.3E+07	2.3E+08
300000	1.8E+11	7.0E+11	8.8E+11	1.7E-01	1.9E-02	1.9E-01	1.8E+08	3.2E+07	2.1E+08
1000000	1.8E+11	3.0E+11	4.8E+11	1.8E-01	8.1E-03	1.9E-01	1.8E+08	2.9E+07	2.1E+08